Dominion Energy Kewaunee, Inc.
5000 Dominion Boulevard, Glen Allen, VA 23060

Dominion®

July 25, 2011

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

Serial No. 11-025
LIC/CDS/R3
Docket No.: 50-305
License No.: DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 244: PROPOSED REVISION TO RADIOLOGICAL ACCIDENT ANALYSIS AND CONTROL ROOM ENVELOPE HABITABILITY TECHNICAL SPECIFICATIONS

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests an amendment to Facility Operating License Number DPR-43 for Kewaunee Power Station (KPS). This proposed amendment would revise the KPS Operating License by modifying the Technical Specifications (TS) and the current licensing basis (CLB) to incorporate changes to the current radiological accident analysis (RAA) of record. This proposed amendment would revise the current RAA for the design-basis accidents (DBAs) described in Chapter 14 of the KPS Updated Safety Analysis Report (USAR).

This amendment would also fulfill a commitment made to the NRC in response to Generic Letter 2003-01, "Control Room Habitability" (references 1 and 2). The commitment stated that DEK would submit proposed changes to the KPS TS based on the final approved version of TSTF-448, "Control Room Habitability."

Attachment 1 to this letter contains a description, safety evaluation, significant hazards consideration determination, and environmental considerations analysis for the proposed changes. Attachment 2 contains marked-up Technical Specification pages. Attachment 3 contains marked-up Technical Specification Bases pages for information. Attachment 4 contains the revised RAA. Attachment 5 contains an evaluation of two proposed new manual actions that are credited in the revised RAA. Enclosed with this submittal is a digital video disc (DVD) containing the meteorological data used to determine the atmospheric dispersion factor (X/Q) values used in the revised RAA.

DEK requests approval of the proposed amendment by July 29, 2012. Once approved, the amendment will be implemented within 90 days.

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

A102
NRR

If you have questions or require additional information, please contact Mr. Craig Sly at 804-273-2784.

Very truly yours,

J. Alan Price
Vice President – Nuclear Engineering

COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

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

Acknowledged before me this 25th day of July, 2011.

My Commission expires: 4/30/2015.

Ginger Lynn Rutherford
NOTARY PUBLIC
Commonwealth of Virginia
Reg. # 310847
My Commission Expires 4/30/2015

Ginger L Rutherford
Notary Public

Attachments:

1. Discussion of Change, Safety Evaluation, Significant Hazards Determination, and Environmental Considerations
2. Marked-Up Operating License and Technical Specifications Pages
3. Marked-Up Technical Specifications Bases Pages
4. Radiological Accident Analysis and Discussion of Associated Technical Specification Changes
5. Evaluation of New Proposed Manual Actions

Enclosure

1. Digital Video Disk (DVD) – Kewaunee Power Station, Meteorological Data, 2002 – 2006, X/Q Calculation Support

Enclosed with this letter is a digital video disk (DVD) which contains this submittal's Enclosure, labeled:

Dominion Energy Kewaunee, Inc.
Kewaunee Power Station
License Amendment Request 244
Proposed Revision to Radiological Accident Analysis and Control Room Envelope Habitability Technical Specifications
Serial No. 11-025
Docket No.: 50-305
License No.: DPR-43

The enclosed DVD contains the following files:

	File Name	Description	File Size	Sensitivity
001	CD Contents.pdf	Description of contents on CD, Met Data file structure, and ARCON 96 RSF and LOG Files	114 KB	publicly available
002	BaseMetData.pdf	2002-2006 Base Met Data	3,510 KB	publicly available
003	PAVANDInputs.txt	PAVAND Input Files	8 KB	publicly available
004	PAVANDInputs.txt	PAVAND Inputs (description/explanation)	175 KB	publicly available
005	PAVANDMetData.pdf	2002-2006 Met data Joint Frequency Distribution PVAND Input	92 KB	publicly available
006	PAVANDOutput.pdf	PAVAND Output File	287 KB	publicly available
007	PAVANDOutput.txt	PAVAND Output File	473 KB	publicly available
008	ARCON96Data.txt	2002-2006 Met Data	1,584 KB	publicly available
	Folder	ARCON96 RSF & LOG Files	180 KB	publicly available

References:

1. Generic Letter 2003-01, "Control Room Habitability," dated June 12, 2003. [ADAMS Accession No. ML031620248]
2. 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]

Commitments made by this letter:

1. 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.
2. 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 (within 30 minutes). 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).
3. DEK will provide the necessary administrative procedures to ensure that in the event of a fuel handling accident inside containment, an open equipment hatch can and will be promptly closed following containment evacuation (within 45 minutes). 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.
4. DEK will relocate the current technical specification requirements related to Radiation Monitor R-23 to the KPS Technical Requirements Manual as part of the implementation of this amendment.

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

Mr. Karl D. Feintuch
NRC Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North, Mail Stop O8-H4A
11555 Rockville Pike
Rockville, MD 20852-2738

NRC Senior Resident Inspector
Kewaunee Power Station

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

ATTACHMENT 1

LICENSE AMENDMENT REQUEST 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

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

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.

<u>Changes to TS 3.7.16, "Secondary Specific Activity"</u>

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 <u>recently</u> 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 "*OR 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

TS 3.4.16, "RCS Specific Activity"

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

[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.

Calculation of DE Xe-133

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.

Incorporate the term "recently" into the APPLICABILITY and Required Action A.1 of TS 3.9.6

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.

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

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.

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[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.

ATTACHMENT 2

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

MARKED-UP OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

Insert 1: Proposed Change to the Kewaunee Operating License

(XX) Upon implementation of Amendment No. [ ] adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by TS SR 3.7.10.4, in accordance with TS 5.5.17.c.1, and the assessment of CRE habitability as required by Specification TS 5.5.17.c.2, shall be considered met. Following implementation:

(a) The first performance of TS 3.7.10.4, in accordance with Specification TS 5.5.17.c.1, shall be within the specified Frequency of 6 years, plus the 18-month allowance of TS SR 3.0.2, as measured from December 15, 2004, the date of the most recent successful tracer gas test, as stated in the April 1, 2005 letter response to Generic Letter 2003-01, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

(b) The first performance of the periodic assessment of CRE habitability; Specification TS 5.5.17.c.2, shall be within 3 years, plus the 9-month allowance of TS SR 3.0.2, as measured from December 15, 2004, the date of the most recent successful tracer gas test, as stated in the April 1, 2005 letter response to Generic Letter 2003-01, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

1.1 Definitions

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

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ------------NOTE------------ Only applicable during movement of irradiated fuel assemblies within containment. ---------------------------------	C.1	Place and maintain containment purge and vent valves in closed position.	Immediately
	OR		
One or more Functions with one or more automatic actuation trains inoperable. OR Two or more radiation monitoring channels inoperable. OR Required Action and associated Completion Time for Condition A not met.	C.2	Enter applicable Conditions and Required Actions of LCO 3.9.6, "Containment Penetrations," for containment purge and vent isolation valves made inoperable by isolation instrumentation.	Immediately

Table 3.3.6-1 (page 1 of 1)
Containment Purge and Vent Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Automatic Actuation Logic and Actuation Relays	1,2,3,4, (a)	2 trains	SR 3.3.6.2
2. Containment Radiation			
a. Gaseous	1,2,3,4, (a)	2	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.4
b. Particulate	1,2,3,4, (a)	1	SR 3.3.6.1 SR 3.3.6.3 SR 3.3.6.4
3. Containment Isolation - Manual Initiation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3.a, for all initiation functions and requirements.		
4. Containment Spray - Manual Initiation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 2.a, for all initiation functions and requirement.		
5. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all functions and requirements.		

(a) During movement of irradiated fuel assemblies within containment.

3.3 INSTRUMENTATION

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

LCO 3.3.7 The CRPAR System actuation instrumentation for each Function in Table 3.3.7-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.7-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Automatic Actuation Logic and Actuation Relay train inoperable.	A.1 Place associated CRPAR train in emergency mode.	7 days
B. Two Automatic Actuation Logic and Actuation Relay trains inoperable. <u>OR</u> Control Room Vent Radiation Monitor inoperable.	B.1.1 Place one CRPAR train in emergency mode.	Immediately
	<u>AND</u>	
	B.1.2 Enter applicable Conditions and Required Actions for one CRPAR train made inoperable by inoperable CRPAR System actuation instrumentation.	Immediately
	<u>OR</u>	
	B.2 Place both CRPAR trains in emergency mode.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time for Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours
D. Required Action and associated Completion Time for Condition A or B not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately
E. Required Action and associated Completion Time for Condition A or B not met in MODE 5 or 6.	E.1 Initiate action to restore one CRPAR train to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----

Refer to Table 3.3.7-1 to determine which SRs apply for each CRPAR System Actuation Function.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.2	Perform COT in accordance with the Setpoint Control Program.	92 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.7.3	-------------------------------NOTE------------------------------ This Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation. --- Perform ACTUATION LOGIC TEST.	18 months
SR 3.3.7.4	Perform CHANNEL CALIBRATION in accordance with the Setpoint Control Program.	18 months

Table 3.3.7-1 (page 1 of 1)
CRPAR System Actuation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Automatic Actuation Logic and Actuation Relays	1, 2, 3, 4, 5, 6, (a)	2 trains	SR 3.3.7.3
2 Control Room Vent Radiation Monitor	1, 2, 3, 4, 5, 6, (a)	1	SR 3.3.7.1 SR 3.3.7.2 SR 3.3.7.4
3. Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

(a) During movement of irradiated fuel assemblies.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 RCS DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 specific activity shall be within limits.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 not within limit.	------------------NOTE------------------ LCO 3.0.4.c is applicable. ---	
	A.1 Verify DOSE EQUIVALENT I-131 ≤ 20 μCi/gm.	Once per 4 hours
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours
B. DOSE EQUIVALENT XE-133 not within limit.	------------------NOTE------------------ LCO 3.0.4.c is applicable. ---	
	B.1 Restore DOSE EQUIVALENT XE-133 to within limit.	48 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. OR DOSE EQUIVALENT I-131 > 20 μCi/gm.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.16.1	Verify reactor coolant DOSE EQUIVALENT XE-133 specific activity ≤ 595 μCi/gm.	7 days
SR 3.4.16.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period

3.7 PLANT SYSTEMS

3.7.10 Control Room Post Accident Recirculation (CRPAR) System

LCO 3.7.10 Two CRPAR trains shall be OPERABLE.

--NOTE--
The control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRPAR train inoperable.	A.1 Restore CRPAR train to OPERABLE status.	7 days
B. Two CRPAR trains inoperable due to inoperable control room boundary in MODE 1, 2, 3, or 4.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1	Place OPERABLE CRPAR train in emergency mode.	Immediately
	OR		
	D.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CRPAR trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	E.1	Suspend movement of irradiated fuel assemblies.	Immediately
F. Two CRPAR trains inoperable in MODE 1, 2, 3, or 4 for reasons other than Condition B.	F.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Operate each CRPAR train for ≥ 15 minutes.	31 days
SR 3.7.10.2	Perform required CRPAR filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.10.3	Verify each CRPAR train actuates on an actual or simulated actuation signal.	18 months

Insert 2:

B. One or more CRPAR trains inoperable due to an inoperable CRE boundary in Modes 1, 2, 3, or 4.	B.1 Initiate action to implement mitigating actions. AND B.2 Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits. AND B.3 Restore CRE boundary to OPERABLE status.	Immediately 24 hours 90 days

Insert 3:

OR Required Actions and associated Completion Times of Condition B not met during movement of recently irradiated fuel assemblies.		

3.7 PLANT SYSTEMS

3.7.11 Control Room Air Conditioning (CRAC) Alternate Cooling System

LCO 3.7.11 Two CRAC Alternate Cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CRAC Alternate Cooling train inoperable.	A.1 Restore CRAC Alternate Cooling train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CRAC Alternate Cooling train in operation. OR C.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately
D. Two CRAC Alternate Cooling trains inoperable during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

3.7 PLANT SYSTEMS

3.7.16 Secondary Specific Activity

LCO 3.7.16 The specific activity of the secondary coolant shall be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limit.	A.1 Be in MODE 3. AND A.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the specific activity of the secondary coolant is ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131.	31 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and

b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A. --	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	A.2.3 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
	B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	AND	
	B.3 Initiate action to restore required DG to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 One DC electrical power subsystem shall be OPERABLE to support one subsystem of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution System - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DC electrical power subsystem inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required DC electrical power subsystem to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 One inverter shall be OPERABLE to support the 120 VAC electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required inverter to OPERABLE status.	Immediately

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

---NOTE---

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC instrument bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	

3.9 REFUELING OPERATIONS

3.9.6 Containment Penetrations

LCO 3.9.6 The containment penetrations shall be in the following status:

a. The equipment hatch is closed and held in place by four bolts;

b. One door in each air lock is capable of being closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere is either:

   1. Closed by a manual or automatic isolation valve, blind flange, or equivalent; or

   2. Capable of being closed by an OPERABLE Containment Purge and Vent Isolation System.

APPLICABILITY: During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend movement of irradiated fuel assemblies within containment.	Immediately

5.5 Programs and Manuals

5.5.16 Setpoint Control Program (continued)

10 CFR 50.90 is required to change the listed value of the NTSP, AV, AFT, and ALT (as applicable) for each Function described in Paragraph a.

c. The program shall establish methods to ensure that Functions described in Paragraph a. will function as required by verifying the as-left and as-found settings are consistent with the list of values established by Paragraph b. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

d. The program shall identify the Functions described in Paragraph a. that are automatic protective devices related to variables having significant safety functions as delineated by 10 CFR 50.36(c)(1)(ii)(A). The NTSP of these Functions are Limiting Safety System Settings. These Functions shall be demonstrated to be functioning as required by applying the following requirements during CHANNEL CALIBRATIONS, CHANNEL OPERATIONAL TESTS, and TRIP ACTUATING DEVICE OPERATIONAL TESTS that verify the NTSP.

   1. The as-found value of the instrument channel trip setting shall be compared with the previous as-left value or the specified NTSP.

   2. If the as-found value of the instrument channel trip setting differs from the previous as-left value or the specified NTSP by more than the pre-defined test acceptance criteria band (i.e., the specified AFT), then the instrument channel shall be evaluated before declaring the SR met and returning the instrument channel to service. This condition shall be entered in the plant corrective action program.

   3. If the as-found value of the instrument channel trip setting is less conservative than the specified AV, then the SR is not met and the instrument channel shall be immediately declared inoperable.

   4. The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the NTSP at the completion of the surveillance test; otherwise, the channel is inoperable (setpoints may be more conservative than the NTSP provided that the as-found and as-left tolerances apply to the actual setpoint used to confirm channel performance).

e. The program shall be specified in the Technical Requirements Manual.

Insert 4:

5.5.17 Control Room Envelope Habitability Program

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

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

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

c. Requirements for:

   1. Determining the unfiltered air in-leakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and;

   2. Assessing CRE habitability at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. 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.

e. The quantitative limits on unfiltered air in-leakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 3.0.2 are applicable to the frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and assessing the CRE boundary as required by TS 5.5.17.c. and TS 5.5.17.d., respectively.

ATTACHMENT 3

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

MARKED-UP TECHNICAL SPECIFICATIONS BASES PAGES

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

B 3.3 INSTRUMENTATION

B 3.3.6 Containment Purge and Vent Isolation Instrumentation

BASES

BACKGROUND

Containment purge and vent isolation instrumentation closes the containment isolation valves in the Containment Vessel Air Handling System, consisting of the Containment Air Cooling and Containment Purge and Vent Systems. This action isolates the containment atmosphere from the environment to minimize releases of radioactivity in the event of an accident. The Containment Air Cooling System may be in use during reactor operation and the Containment Purge and Vent System will be in use with the reactor shutdown.

Containment purge and vent isolation initiates on an automatic safety injection (SI) signal; a manual SI signal; a manual containment vent isolation signal; or a manual containment spray signal (of both trains). The Bases for LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," discuss these modes of initiation.

Three radiation monitoring channels are also provided as input to the containment purge and vent isolation. The three channels measure containment radiation at two locations. One channel is a particulate monitor (R-11), the second channel is a radioactive gas monitor (R-12), and the third channel is also a radioactive gas monitor (R-21). The three channels are separated into two trains with channel R-21 designated as Train A and channels R-11 and R-12 designated as Train B. All three detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purpose of this LCO the three channels are not considered redundant. Since the radiation monitors constitute a sampling system, various components such as sample line valves, sample line heaters, sample pumps, and filter motors are required to support monitor OPERABILITY.

Each of the purge systems has inner and outer containment isolation valves in its supply and exhaust ducts. A high radiation signal from any one of the three channels initiates containment purge isolation, which closes both inner and outer containment isolation valves in the Containment Purge and Vent System and the 2 inch containment vent isolation valves. These valves are described in the Bases for LCO 3.6.3, "Containment Isolation Valves."

BASES

APPLICABLE SAFETY ANALYSES

The safety analyses assume that the containment remains intact with penetrations unnecessary for core cooling isolated early in the event. The isolation of the purge valves has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed. The containment purge and vent isolation radiation monitors act as backup to the SI signal to ensure closing of the purge and vent valves. They are also the primary means for automatically isolating containment in the event of a fuel handling accident during shutdown. Containment isolation contributes to both meeting the containment leakage rate assumptions of the safety analyses and ensuring that the calculated control room and accidental offsite radiological doses are below 10 CFR 50.67 (Ref. 1) limits. Due to radioactive decay, containment is only required to isolate during fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

The containment purge and vent isolation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that the instrumentation necessary to initiate Containment Purge and Vent Isolation, listed in Table 3.3.6-1, is OPERABLE.

1. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains of Automatic Actuation Logic and Actuation Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

   Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b, SI, and ESFAS Function 3, Containment Isolation. The applicable MODES and specified conditions for the containment purge and vent isolation portion of these Functions are different than those for their Containment isolation and SI roles. If one or more of the SI or Containment isolation Functions becomes inoperable in such a manner that only the Containment Purge and Vent Isolation Function is affected, the Conditions applicable to their SI and Containment isolation Functions need not be entered. The less restrictive Actions specified for inoperability of the Containment Purge and Vent Isolation Functions specify sufficient compensatory measures for this case.

2. Containment Radiation

   The LCO specifies three required channels of radiation monitors to ensure that the radiation monitoring instrumentation necessary to initiate Containment Purge and Vent Isolation remains OPERABLE.

For sampling systems, channel OPERABILITY involves more than OPERABILITY of the channel electronics. OPERABILITY will also

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LCO (continued)

require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, since these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.

The radioactive gas monitor (R-21) has two flow path alignments; it can be aligned to the 36 inch containment purge exhaust line or to the containment atmosphere via the same penetration used by particulate monitor R-11 and radioactive gas monitor R-12. However, since the 36 inch containment purge exhaust line is isolated and sealed in MODES 1, 2, 3, and 4, for the radioactive gas monitor R-21 to be OPERABLE, it must be aligned to the containment atmosphere via the same containment penetration as the R-11 and R-12 radiation monitors.

3. Containment Isolation - Manual Initiation

Refer to LCO 3.3.2, Function 3.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

4. Containment Spray - Manual Initiation

Refer to LCO 3.3.2, Function 2.a, for all initiating Functions and requirements. This Function provides the manual initiation capability for containment ventilation isolation.

5. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements. This Function provides both manual and automatic initiation capability for containment ventilation isolation.

APPLICABILITY

The Automatic Actuation Logic and Actuation Relays and Containment Radiation Functions are required OPERABLE in MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) within containment. Under these conditions, the potential exists for a release of fission product radioactivity into containment. Therefore, the containment purge and vent isolation instrumentation must be OPERABLE in these MODES.

While in MODES 5 and 6, the containment purge and vent isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to

BASES

APPLICABILITY (continued)

ensure post accident offsite doses are maintained within the limits of Reference 1.

The Applicability for the containment purge and vent isolation on the Containment Isolation - Manual Initiation, Containment Spray - Manual Initiation, and Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Containment Isolation - Manual Initiation, Containment Spray - Manual Initiation, and Safety Injection Function Applicability.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.6-1. The Completion Time(s) of the inoperable channel(s)/train(s) of a Function will be tracked separately for each Function starting from the time the Condition was entered for that Function.

<u>A.1</u>

Condition A applies to the failure of one containment radiation monitor channel. Since the three containment radiation monitors measure different parameters, failure of a single channel may result in loss of the radiation monitoring Function for certain events. Consequently, the failed channel must be restored to OPERABLE status. The 4 hours allowed to restore the affected channel is justified by the low likelihood of events occurring during this interval, and recognition that one or more of the remaining channels will respond to most events.

<u>B.1</u>

Condition B applies to one or more Automatic Actuation Logic and Actuation Relays trains and addresses the train orientation of the master and slave relays for these Functions. It also addresses the failure of

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ACTIONS

B.1 (continued)

multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1.

If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action for the applicable Conditions of LCO 3.6.3 is met for each valve made inoperable by failure of isolation instrumentation.

A Note is added stating that Condition B is only applicable in MODE 1, 2, 3, or 4.

C.1 and C.2

Condition C applies to one or more Automatic Actuation Logic and Actuation Relays trains and addresses the train orientation of the master and slave relays for these Functions. It also addresses the failure of multiple radiation monitoring channels, or the inability to restore a single failed channel to OPERABLE status in the time allowed for Required Action A.1. If a train is inoperable, multiple channels are inoperable, or the Required Action and associated Completion Time of Condition A are not met, operation may continue as long as the Required Action to place and maintain containment purge and vent isolation valves in their closed position is met or the applicable Conditions of LCO 3.9.6, "Containment Penetrations," are met for each valve made inoperable by failure of isolation instrumentation. The Completion Time for these Required Actions is Immediately.

A Note states that Condition C is applicable during movement of recently irradiated fuel assemblies within containment.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.6-1 determines which SRs apply to which Containment Purge and Vent Isolation Functions.

SR 3.3.6.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.6.2

SR 3.3.6.2 is the performance of an ACTUATION LOGIC TEST. The train being tested is placed in the test condition, thus preventing inadvertent actuation. All possible logic combinations are tested for each protection function. This test is performed every 92 days on a STAGGERED TEST BASIS. The Surveillance interval is justified in Reference 2.

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

SR 3.3.6.3

A COT is performed every 92 days on each required channel to ensure the entire channel will perform the intended Function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.3 (continued)

because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Frequency is based on the staff recommendation for increasing the availability of radiation monitors according to NUREG-1366 (Ref. 3). This test verifies the capability of the instrumentation to provide the containment purge and vent system isolation. The Setpoint Control Program (SCP) has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

SR 3.3.6.4

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. The SCP has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.

The Frequency is based on operating experience and is consistent with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50.67.
2. WCAP-15376, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," October 2000.
3. NUREG-1366, December 1992.

B 3.3 INSTRUMENTATION

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

BASES

BACKGROUND

The CRPAR System provides an enclosed control room environment from which the unit can be operated following an uncontrolled release of radioactivity. The CRPAR System is part of the Control Room Air Conditioning System. During normal unit operation, the Control Room Air Conditioning System provides cooling and heating of recirculated and fresh air to ventilate the control room. Upon receipt of an actuation signal, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. This system is described in the Bases for LCO 3.7.10, "Control Room Post Accident Recirculation (CRPAR) System."

~~The actuation instrumentation 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 fan(s) by manual switches in the control room.~~ The CRPAR System is ~~also~~ actuated by a safety injection (SI) signal. The control room operator can also start the CRPAR fan(s) by manual switches in the control room. The SI Function is discussed in LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

APPLICABLE SAFETY ANALYSES

The control room must be kept habitable for the operators stationed there during accident recovery and post accident operations.

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~The radiation monitor~~Manual actuation of the CRPAR System ~~in MODES 5 and 6,~~ and a requirement for the control room envelope to be isolated during movement of recently irradiated fuel assemblies (TS 3.7.10) is the primary means to ensure control room habitability in the event of a fuel handling~~, volume control tank, or waste gas decay tank rupture~~ accident.

The CRPAR System actuation instrumentation satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requirements ensure that instrumentation necessary to initiate the CRPAR System is OPERABLE.

1. Automatic Actuation Logic and Actuation Relays

   The LCO requires two trains of Actuation Logic and Relays OPERABLE to ensure that no single random failure can prevent automatic actuation.

   Automatic Actuation Logic and Actuation Relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b., SI, in LCO 3.3.2 and include the slave relays that send the SI signal to the CRPAR System. The applicable MODES and specified conditions for the CRPAR System portion of these functions are different than those specified for their SI roles. If one or more of the SI functions becomes inoperable in such a manner that only the CRPAR System function is affected, the Conditions applicable to their SI function need not be entered. The less restrictive Actions specified for inoperability of the CRPAR System Functions specify sufficient compensatory measures for this case.

~~2. Control Room Vent Radiation Monitor~~

   ~~The LCO specifies one Control Room Vent Radiation Monitor to ensure that the radiation monitoring instrumentation necessary to initiate the CRPAR System remains OPERABLE.~~

   ~~For sampling systems, channel OPERABILITY involves more than OPERABILITY of channel electronics. OPERABILITY may also require correct valve lineups, sample pump operation, and filter motor operation, as well as detector OPERABILITY, if these supporting features are necessary for trip to occur under the conditions assumed by the safety analyses.~~

3. Safety Injection

Refer to LCO 3.3.2, Function 1, for all initiating Functions and requirements.

BASES

APPLICABILITY

The CRPAR Functions must be OPERABLE in MODES 1, 2, 3, 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of the critical reactor core within the previous 375 hours). ~~The Functions must also be OPERABLE in MODES 5 and 6 when required for a waste gas decay tank rupture accident, to ensure a habitable environment for the control room operators.~~

The Applicability for the CRPAR actuation on the ESFAS Safety Injection Functions are specified in LCO 3.3.2. Refer to the Bases for LCO 3.3.2 for discussion of the Safety Injection Function Applicability.

ACTIONS

The most common cause of channel inoperability is outright failure or drift of the bistable or process module sufficient to exceed the tolerance allowed by the unit specific calibration procedures. Typically, the drift is found to be small and results in a delay of actuation rather than a total loss of function. This determination is generally made during the performance of a COT, when the process instrumentation is set up for adjustment to bring it within specification. If the Trip Setpoint is less conservative than the tolerance specified by the calibration procedure, the channel must be declared inoperable immediately and the appropriate Condition entered.

A.1

Condition A applies to the Automatic Actuation Logic and Actuation Relays Function of the CRPAR System.

If one train is inoperable, 7 days are permitted to restore it to OPERABLE status. The 7 day Completion Time is the same as is allowed if one train of the mechanical portion of the system is inoperable. The basis for this Completion Time is the same as provided in LCO 3.7.10. If the train cannot be restored to OPERABLE status, the associated CRPAR train must be placed in the emergency mode of operation. This accomplishes the actuation instrumentation Function and places the unit in a conservative mode of operation.

B.1.1, B.1.2, and B.2

Condition B applies to the failure of two Automatic Actuation Logic and Actuation Relay trains ~~or the Control Room Vent Radiation Monitor~~. The first Required Action is to place one CRPAR train in the emergency mode of operation immediately. This accomplishes the actuation instrumentation Function that may have been lost and places the unit in a conservative mode of operation. The applicable Conditions and Required

BASES

ACTIONS

B.1.1, B.1.2, and B.2 (continued)

Actions of LCO 3.7.10 must also be entered for the CRPAR train made inoperable by the inoperable actuation instrumentation and not placed in the emergency mode of operation. This ensures appropriate limits are placed upon train inoperability as discussed in the Bases for LCO 3.7.10.

Alternatively, both CRPAR trains may be placed in the emergency mode. This ensures the CRPAR function is performed even in the presence of a single failure.

C.1 and C.2

Condition C applies when the Required Action and associated Completion Time for Condition A or B have not been met and the unit is in MODE 1, 2, 3, or 4. The unit must be brought to a MODE in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

Condition D applies when the Required Action and associated Completion Time for Condition A or B have not been met when recently irradiated fuel assemblies are being moved. Movement of recently irradiated fuel assemblies must be suspended immediately to reduce the risk of accidents that would require CRPAR System actuation.

~~E.1~~

~~Condition E applies when the Required Action and associated Completion Time for Condition A or B have not been met in MODE 5 or 6. Actions must be initiated to restore the inoperable train(s) to OPERABLE status immediately to ensure adequate isolation capability in the event of a waste gas decay tank rupture.~~

BASES

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that Table 3.3.7-1 determines which SRs apply to which CRPAR System Actuation Functions.

~~SR 3.3.7.1~~

~~Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.~~

~~Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.~~

~~The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.~~

~~SR 3.3.7.2~~

~~A COT is performed once every 92 days on each required channel to ensure the entire channel will perform the intended function. This test verifies the capability of the instrumentation to provide the CRPAR System actuation. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable COT of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The Setpoint Control Program (SCP) has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology. The Frequency is based on the known reliability of the monitoring equipment and has been shown to be acceptable through operating experience.~~

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.7.~~3~~1

SR 3.3.7.3 is the performance of an ACTUATION LOGIC TEST. For the portion of the logic common to ESFAS, Function 1.b ACTUATION LOGIC TEST, the train being tested is placed in the test condition, thus preventing inadvertent actuation and all possible SI logic combinations are tested for each protection function. For the portion of the logic not tested as part of the ESFAS Function 1.b ACTUATION LOGIC TEST (i.e., the slave relay), actuation of the end devices may occur. The Frequency of 18 months is based on the refueling outage cycle, since the slave relay cannot be tested at power without resulting in actuation of affected components.

The SR is modified by a Note stating that the Surveillance is only applicable to the actuation logic of the ESFAS Instrumentation.

~~SR 3.3.7.4~~

~~A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.~~

~~The SCP has controls which require verification that the instrument channel functions as required by verifying the as-left and as-found setting are consistent with those established by the setpoint methodology.~~

~~The Frequency is based on operating experience and is consistent with the typical industry refueling cycle~~.

REFERENCES

1. WCAP-15376, Rev. 0, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," October 2000.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

USAR General Design Criteria (GDC) 16 (Ref. 1) states that means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. USAR, Section 6.5 (Ref. 2) describes the capabilities of the leakage monitoring indication systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from the steam generators (SGs) is 150 gallons per day per SG. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is a condition assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. Other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR), locked reactor coolant pump rotor, and control rod ejection. The primary to secondary leakage contaminates the secondary fluid.

The radiological accident analysis ~~(Ref. 3)~~ for SGTR assumes the contaminated secondary fluid is released to the environment from the ruptured and the intact SGs. The release from the ruptured SG occurs until ~~30~~ 55 minutes after the reactor trip and the release from the intact SG occurs until ~~24~~ 29 hours after the reactor trip when residual heat removal is placed in service. The 150 gallons per day SG primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

~~The SLB is less limiting for site radiation releases.~~ The safety analysis for the SLB accident assumes the 150 gallons per day primary to secondary LEAKAGE is through the affected generator as an initial condition. The dose consequences resulting from the SLB accident are ~~well~~ within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., ~~a small fraction of these~~ RG 1.183, Rev 0 limits).

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

APPLICABILITY (continued)

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.2 (continued)

operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

The Surveillance is modified by a Note which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

REFERENCES

1. USAR, Section 4.1.3.2, GDC 16, "Monitoring Reactor Coolant Leakage."

2. USAR, Section 6.5, Leakage Detection and Provisions for the Primary and Auxiliary Coolant Loops.

3. ~~Westinghouse calculation CN-CRA-99-36, Steam Generator Tube Rupture.~~ Not Used

4. NEI 97-06, "Steam Generator Program Guidelines."

5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

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 (Ref. 1). Doses to control room operators must be limited per 10 CFR 50, GDC 19. The limits on specific activity ensure that the offsite and control room doses are appropriately limited during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO 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.

The LCO 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 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting offsite and control room doses meet the appropriate RG 1.183 acceptance criteria following a MSLB or SGTR accident. The safety analyses (Refs. 3 and 4) assume the specific activity of the reactor coolant is at the LCO limits and an existing reactor coolant steam generator (SG) tube leakage rate of 150 gallons per day exists. The safety analyses assume the specific activity of the secondary coolant is at its limit of ~~0.1~~ 0.05 μCi/gm DOSE EQUIVALENT I-131 from LCO ~~3.7.17~~ 3.7.16, "Secondary Specific Activity."

The analyses for the MSLB and SGTR accidents establish the acceptance limits for RCS specific activity. Reference to these analyses is used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

The safety analyses consider two cases of reactor coolant iodine specific activity. One case assumes specific activity at ~~1.0~~ 0.1 μCi/gm DOSE EQUIVALENT I-131 with an accident initiated iodine spike that increases the rate of release of iodine from the fuel rods containing cladding defects to the primary coolant immediately after a MSLB ~~or SGTR~~ (by a factor of 500) or a SGTR (by a factor of 335). The second case assumes the initial reactor coolant iodine activity at ~~20.0~~ 10.0 μCi/gm DOSE EQUIVALENT I-131 for the MSLB accident and ~~20.0~~ 10.0 μCi/gm DOSE EQUIVALENT I-131 for the SGTR accident due

BASES

APPLICABLE SAFETY ANALYSES (continued)

to an iodine spike caused by a reactor or an RCS transient prior to the accident. In both cases, the noble gas specific activity is assumed to be ~~595~~ 16.4 μCi/gm DOSE EQUIVALENT XE-133.

The SGTR analysis also considers a possible loss of offsite power at the same time as the reactor trip. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature ΔT signal.

A coincident loss of offsite power would cause the steam dump valves to close to protect the condenser. The rise in pressure in the ruptured SG is assumed to discharge radioactively contaminated steam to the atmosphere through the ~~main steam safety valves~~ SG power operated relief valves. The unaffected SG removes core decay heat by venting steam to the atmosphere until the event is terminated.

The MSLB radiological analysis assumes that offsite power is lost at the same time as the pipe break occurs outside containment. Reactor trip occurs after the generation of an SI signal on low steam line pressure. The affected SG blows down completely and steam is vented directly to the atmosphere. The unaffected SG removes core decay heat by venting steam to the atmosphere until the cooldown ends and the RHR System is placed in service.

Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed ~~20.0~~ 10.0 μCi/gm for more than 48 hours.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The iodine specific activity in the reactor coolant is limited to ~~1.0~~ 0.1 μCi/gm DOSE EQUIVALENT I-131, and the noble gas specific activity in the reactor coolant is limited to ~~595~~ 16.4 μCi/gm DOSE EQUIVALENT XE-133. The limits on specific activity ensure that offsite and control room doses will meet the appropriate RG 1.183 acceptance criteria (Ref. 2)

The MSLB and SGTR accident analyses (Refs. 3 and 4) show that the calculated doses are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of a MSLB or SGTR, lead to doses that exceed the RG 1.183 acceptance criteria (Ref. 2).

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, operation within the LCO limits for DOSE EQUIVALENT I-131 and DOSE EQUIVALENT XE-133 is necessary to limit the potential consequences of a MSLB or SGTR to within the RG 1.183 acceptance criteria (Ref. 2).

In MODES 5 and 6, the steam generators are not normally being used for decay heat removal, the RCS and steam generators are depressurized, and primary to secondary leakage is minimal. Therefore, the monitoring of RCS specific activity is not required.

ACTIONS

A.1 and A.2

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the specific activity is ≤ ~~20~~ 10 μCi/gm. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is continued every 4 hours to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limit within 48 hours. The Completion Time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) relying on Required Actions A.1 and A.2 while the DOSE EQUIVALENT I-131 LCO limit is not met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

B.1

With the DOSE EQUIVALENT XE-133 greater than the LCO limit, DOSE EQUIVALENT XE-133 must be restored to within limit within 48 hours. The allowed Completion Time of 48 hours is acceptable since it is expected that, if there were a noble gas spike, the normal coolant noble gas concentration would be restored within this time period. Also, there is a low probability of a MSLB or SGTR occurring during this time period.

BASES

ACTIONS

B.1 (continued)

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S), relying on Required Action B.1 while the DOSE EQUIVALENT XE-133 LCO limit is met. This allowance is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient-specific activity excursions while the plant remains at, or proceeds to, power operation.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met, or if the DOSE EQUIVALENT I-131 is > ~~20~~ 10 μCi/gm, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the noble gas specific activity of the reactor coolant at least once every 7 days. This measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in noble gas specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The 7 day Frequency considers the low probability of a gross fuel failure during the time.

Due to the inherent difficulty in detecting Kr-85 in a reactor coolant sample due to masking from radioisotopes with similar decay energies, such as F-18 and I-134, it is acceptable to include the minimum detectable activity for Kr-85 in the SR 3.4.16.1 calculation. If a specific noble gas nuclide listed in the definition of DOSE EQUIVALENT XE-133 is not detected, it should be assumed to be present at the minimum detectable activity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed to ensure iodine specific activity remains within the LCO limit during operation and following fast power changes when iodine spiking is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering noble gas activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following iodine spike initiation; samples at other times would provide inaccurate results.

REFERENCES

1. 10 CFR 50.67.
2. Regulatory Guide 1.183, July 2000.
3. USAR, Section 14.2.4.
4. USAR, Section 14.2.5.

B 3.7 PLANT SYSTEMS

B 3.7.10 Control Room Post Accident Recirculation (CRPAR) System

BASES

BACKGROUND

The CRPAR System provides a protected environment from which ~~operators~~ occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or ~~toxic gas~~smoke.

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

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CRPAR System is an emergency system, which is normally in the standby mode of operation. The CRPAR System is part of the Control Room Air Conditioning (CRAC) System. During normal unit operation, the CRAC System provides cooling of recirculated and fresh air to ventilate the control room. Upon receipt of the actuating signal(s), normal outside air intake supply to the ~~control room~~CRE is isolated, both CRPAR fans are started, the flow path through the Emergency Filtration System is opened, and a portion of the return air volume is filtered to remove airborne contaminants and airborne radioactivity, then mixed with the recirculated return air. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

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

The CRPAR System operation in maintaining a habitable environment in the CRE ~~control room habitable~~ is discussed in the USAR, Section 9.6.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers of the CRAC Alternate Cooling System provide double/redundant isolation capability so that the failure of one damper to shut will not result in a breach of control room ventilation isolation. The CRPAR System is designed in accordance with Seismic Category I requirements.

The manual actuation of the CRPAR System during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) is the primary means to ensure CRE habitability in the event of a fuel handling accident while handling recently irradiated fuel. Actuation of the CRPAR System and control room isolation are performed by a SI actuation signal, either automatically or manually initiated. Calculated doses to CRE occupants from a volume control tank rupture or waste gas decay tank rupture are sufficiently small that manual actuation of the CRPAR System is not required for these postulated accidents.

BASES

BACKGROUND (continued)

The CRPAR System is designed to maintain a habitable environment in the CRE ~~control room environment~~ for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem total effective dose equivalent (TEDE).

APPLICABLE SAFETY ANALYSES

The CRPAR System components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the ~~control room envelope~~CRE ensures an adequate supply of filtered air to all areas requiring access. The CRPAR System provides airborne radiological protection for the ~~control room operators~~CRE occupants, as demonstrated by the ~~control room accident~~CRE occupant dose analyses for the most limiting design basis ~~loss of coolant~~ accident~~,~~ fission product release presented in the USAR, Chapter 14 (Ref. 2).

The CRPAR System also provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 6). The evaluation of a smoke challenge also demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panel (Ref. 7).~~for the control room operators in the remote possibility of a fire in the control room, as described in Reference 1.~~

The worst case single active failure of a component of the CRPAR System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CRPAR System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant CRPAR trains are required to be OPERABLE to ensure that at least one is available ~~assuming~~ if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem TEDE to the control room operator in the event of a large radioactive release.

~~The~~ Each CRPAR ~~System~~ train is considered OPERABLE when the individual components necessary to limit ~~operator~~ CRE occupant exposure are OPERABLE~~ in both trains~~. A CRPAR train is OPERABLE when the associated:

a. Fan is OPERABLE;

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions; and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In addition, the CRAC fan in the same train must be OPERABLE when the CRPAR train is required. ~~Furthermore, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors.~~

In order for the CRPAR trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

BASES

LCO (continued)

The LCO is modified by ~~a~~ two Notes. The first Note allows~~allowing~~ the ~~control room~~CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, dampers, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE~~control room~~. This individual will have a method to rapidly close the opening and restore the CRE boundary to a condition equivalent to the design condition when a need for ~~control room~~CRE isolation is indicated.

The second Note requires that the CRE be isolated during movement of recently irradiated fuel assemblies. The fuel handling accident analysis assumes the control room is isolated at the initiation of the accident. Pre-isolation of the control room minimizes infiltration of radioactive materials into the CRE prior to initiation of the CRPAR in the emergency mode and ensures dose to CRE occupants remains within applicable limits.

APPLICABILITY

In MODES 1, 2, 3, and 4, ~~5, and 6,~~ and during movement of recently irradiated fuel assemblies, the CRPAR System must be OPERABLE to ensure that the CRE will remain habitable ~~control operator exposure~~ during and following a DBA.

~~In MODE 5 or 6, the CRPAR System is required to cope with the release from the rupture of an inside waste gas tank.~~

During movement of recently irradiated fuel assemblies, the CRPAR System must be OPERABLE to cope with the release from a fuel handling accident involving handling of recently irradiated fuel. The CRPAR is only required to be OPERABLE during fuel handling involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours), due to radioactive decay.

ACTIONS

A.1

When one CRPAR train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this condition, the remaining OPERABLE CRPAR train is adequate to perform the ~~control room~~CRE occupant protection function. However, the overall reliability is reduced because a ~~single active~~ failure in the OPERABLE CRPAR train could result in loss of CRPAR function. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

B.1, B.2, and B.3

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem TEDE), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24-hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90-day Completion Time is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day Completion Time is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

~~If the control room boundary is inoperable in MODE 1, 2, 3, or 4, the CRPAR trains cannot perform their intended functions. Action must be taken to restore an OPERABLE control room boundary within 24 hours. During the period that the control room boundary is inoperable, appropriate compensatory measures (consistent with the intent of GDC 19) should be utilized to protect control room operators from potential hazards such as radioactive contamination, toxic chemicals, smoke, temperature and relative humidity, and physical security. Preplanned measures should be available to address these concerns for~~

BASES

ACTIONS (continued)

~~intentional and unintentional entry into the condition. The 24 hour Completion Time is reasonable based on the low probability of a DBA occurring during this time period, and the use of compensatory measures. The 24 hour Completion Time is a typically reasonable time to diagnose, plan and possibly repair, and test most problems with the control room boundary.~~

C.1 and C.2

In MODE 1, 2, 3, or 4, if the inoperable CRPAR train or ~~control room~~ the CRE boundary cannot be restored to OPERABLE status within the required Completion Time, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

~~In MODE 5 or 6, or d~~During movement of recently irradiated fuel assemblies, if the inoperable CRPAR train cannot be restored to OPERABLE status within the required Completion Time, action must be taken to immediately place the OPERABLE CRPAR train in the emergency mode. This action ensures that the remaining train is OPERABLE and that any active failure would be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the ~~control room~~CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

E.1

~~In MODE 5 or 6, or d~~During movement of recently irradiated fuel assemblies, with two CRPAR trains inoperable, or with one or more CRPAR trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might ~~enter~~ require isolation of the CRE~~control room~~. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

BASES

ACTIONS (continued)

F.1

If both CRPAR trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable ~~control room~~CRE boundary (i.e., Condition B), the CRPAR System may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.7.10.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. Operating each CRPAR train for ≥ 15 minutes demonstrates the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy~~ availability~~.

SR 3.7.10.2

This SR verifies that the required CRPAR testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP.

SR 3.7.10.3

This SR verifies that each CRPAR train starts and operates on an actual or simulated actuation (~~high radiation and~~ safety injection) signal. The frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle. ~~Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.~~

SR 3.7.10.4

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Condition B must be entered. Required Action B.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 4) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 5). These compensatory measures may also be used as mitigating actions as required by Required Action B.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 3). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

1. USAR, Section 9.6.4.
2. USAR, Chapter 14.
3. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI), "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability," dated January 30, 2004. [ADAMS Accession No. ML040300694].
4. Regulatory Guide 1.196, Rev. 2.
5. NEI 99-03, "Control Room Habitability Assessment," March 2003.
6. Letter from C. R. Steinhardt to NRC, "Submittal of Kewaunee's Updated Control Room Habitability Evaluation Report to Address Concerns Over Control Room Ventilation," dated February 28, 1989.
7. USAR Section 9.6.4.

B 3.7 PLANT SYSTEMS

B 3.7.11 Control Room Air Conditioning (CRAC) Alternate Cooling System

BASES

BACKGROUND

The CRAC Alternate Cooling System provides temperature control for the control room following isolation of the control room during a design basis accident.

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

The CRAC Alternate Cooling System is an emergency system, parts of which also operate during normal unit operations. A single train will provide the required temperature control to maintain the control room between 60° and 85°F during normal operation using the non-safety related chiller. Under accident conditions (i.e., the non-safety related chillers not in service), cooling from the service water aligned directly to the AHU cooling coils will maintain temperature habitability of the control room environment and will maintain environment temperature for equipment operation. With a service water temperature of 80°F and a 95°F air ambient temperature, each CRAC Alternate Cooling train can maintain control room air temperature within the 110°F design temperature limit. The CRAC Alternate Cooling System operation in maintaining the control room temperature is discussed in the USAR, Section 9.6.4 (Ref. 1).

APPLICABLE SAFETY ANALYSES

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

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The CRAC Alternate Cooling System satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two independent and redundant trains of the CRAC Alternate Cooling System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disabling the other train. Total system failure could result in the equipment operating temperature exceeding limits in the event of an accident.

The CRAC Alternate Cooling System is considered to be OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both trains. These components include the cooling coils (with cooling water from the Service Water System) and associated temperature control instrumentation. In addition, the CRAC Alternate Cooling System must be OPERABLE to the extent that air circulation can be maintained.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours), the CRAC Alternate Cooling System must be OPERABLE to ensure that the control room temperature will not exceed equipment operational requirements following isolation of the control room.

In MODE 5 or 6, CRAC Alternate Cooling System is not required for the mitigation of a postulated event.

ACTIONS

A.1

With one CRAC Alternate Cooling train inoperable, action must be taken to restore OPERABLE status within 30 days. In this condition, the remaining OPERABLE CRAC Alternate Cooling train is adequate to maintain the control room temperature within limits. However, the overall reliability is reduced because a single failure in the OPERABLE CRAC Alternate Cooling train could result in loss of CRAC Alternate Cooling System function. The 30 day Completion Time is based on the low probability of an event requiring control room isolation, the consideration that the remaining train can provide the required protection, and that alternate safety or nonsafety related cooling means are available.

B.1 and B.2

In MODE 1, 2, 3, or 4, if the inoperable CRAC Alternate Cooling train cannot be restored to OPERABLE status within the required Completion Time of Condition A, the unit must be placed in a MODE that minimizes the risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed

BASES

ACTIONS

B.1 and B.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1 and C.2

During movement of recently irradiated fuel, if the inoperable CRAC Alternate Cooling train cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CRAC Alternate Cooling train must be placed in operation immediately. This action ensures that the remaining train is OPERABLE and that active failures will be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room (Required Action C.2). This places the unit in a condition that minimizes accident risk. This does not preclude the movement of fuel to a safe position.

D.1

During movement of recently irradiated fuel assemblies, with two CRAC Alternate Cooling trains inoperable, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. This does not preclude the movement of fuel to a safe position.

E.1

If both CRAC Alternate Cooling trains are inoperable in MODE 1, 2, 3, or 4, the CRAC Alternate Cooling System may not be capable of performing its intended function. Therefore, LCO 3.0.3 must be entered immediately.

BASES ~~(continued)~~

SURVEILLANCE REQUIREMENTS	SR 3.7.11.1 This SR verifies that the heat removal capability of the system is sufficient to remove the heat load assumed in the safety analyses in the control room. This SR consists of a combination of testing both redundant cooling units, verifying the availability of cooling water, and calculations. The 18 month Frequency is appropriate since significant degradation of the CRAC Alternate Cooling System is slow and is not expected over this time period.
REFERENCES	1. USAR, Section 9.6.4.

B 3.7 PLANT SYSTEMS

B 3.7.16 Secondary Specific Activity

BASES

BACKGROUND

Activity in the secondary coolant results from steam generator tube outleakage from the Reactor Coolant System (RCS). Under steady state conditions, the activity is primarily iodines with relatively short half lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.13, "RCS Operational LEAKAGE") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.16, "RCS Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half lives (i.e., < 20 hours).

~~With the specified activity limit, the resultant 2 hour total effective dose equivalent (TEDE) dose to a person at the exclusion area boundary (EAB) would be about 0.03 rem if the steam generator power-operated relief valves (PORVs) open for 2 hours following a trip from full power.~~

Operating a unit at the allowable limits could result in a ~~2 hour~~ EAB exposure of a small fraction of the 10 CFR 50.67 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the USAR, Chapter 14 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of ~~0.10~~ 0.05 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit EAB limits (Ref. 1) for TEDE dose ~~rates~~.

BASES

APPLICABLE SAFETY ANALYSES (continued)

With the loss of offsite power, the remaining steam generator is available for core decay heat dissipation by venting steam to the atmosphere through the main steam safety valves (MSSVs) and steam generator PORVs. The Auxiliary Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal (RHR) System to be placed in service. The RHR System then continues to cooldown to 212°F, at which point the release is terminated.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and PORVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ ~~0.10~~ 0.05 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

In MODES 5 and 6, the steam generators are not normally being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

BASES

ACTIONS	A.1 and A.2 DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.
SURVEILLANCE REQUIREMENTS	SR 3.7.16.1 This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31 day Frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.
REFERENCES	1. 10 CFR 50.67. 2. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources - Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the minimum AC sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

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

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

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving recently irradiated fuel. Due to radioactive decay, AC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

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

During MODES 1, 2, 3, and 4, various deviations from the analysis assumptions and design requirements are allowed within the Required Actions. This allowance is in recognition that certain testing and maintenance activities must be conducted provided an acceptable level of risk is not exceeded. During MODES 5 and 6, performance of a significant number of required testing and maintenance activities is also required. In MODES 5 and 6, the activities are generally planned and administratively controlled. Relaxations from MODE 1, 2, 3, and 4 LCO requirements are acceptable during shutdown modes based on:

BASES

APPLICABLE SAFETY ANALYSES (continued)

a. The fact that time in an outage is limited. This is a risk prudent goal as well as a utility economic consideration.

b. Requiring appropriate compensatory measures for certain conditions. These may include administrative controls, reliance on systems that do not necessarily meet typical design requirements applied to systems credited in operating MODE analyses, or both.

c. Prudent utility consideration of the risk associated with multiple activities that could affect multiple systems.

d. Maintaining, to the extent practical, the ability to perform required functions (even if not meeting MODE 1, 2, 3, and 4 OPERABILITY requirements) with systems assumed to function during an event.

This LCO ensures the capability to support systems necessary to avoid immediate difficulty, assuming either a loss of all offsite power or a loss of all onsite diesel generator (DG) power.

AC Sources - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One offsite circuit capable of supplying the onsite Class 1E power distribution subsystem(s) of LCO 3.8.10, "Distribution Systems - Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a distribution system train required to be OPERABLE by LCO 3.8.10, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Together, OPERABILITY of the required offsite circuit and DG ensures the availability of sufficient AC sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel).

The qualified offsite circuit must be capable of maintaining rated frequency and voltage, and accepting required loads during an accident, while connected to the Engineered Safety Feature (ESF) bus(es).

One qualified offsite circuit consists of the 138/4.16 kV Reserve Auxiliary Transformer, powered by the 138 kV portion of the Kewaunee Substation and normally supplying power to Bus 1-6. The other qualified offsite circuit consists of the 13.8 kV tertiary winding of the 345/138 kV Auto Transformer, powered by either the 345 kV or 138 kV portion of the Kewaunee Substation, to the 13.8/4.16 kV Tertiary Auxiliary Transformer normally supplying power to Bus 1-5. The offsite circuits also include the supply breakers to buses 1-5 and 1-6. While each circuit has connections to each 4.16 kV bus, each circuit is only required to be capable of

BASES

LCO (continued)

supplying one of the 4.16 kV buses at a time. However, if only one offsite circuit is used to meet the LCO requirement, then it must be supplying both buses 1-5 and 1-6.

The DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage. This sequence must be accomplished within 10 seconds. The DG must be capable of accepting required loads within the assumed loading sequence intervals, and continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as DG in standby with the engine hot and DG in standby at ambient conditions.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

It is acceptable for trains to be cross tied during shutdown conditions, allowing a single offsite power circuit to supply all required trains.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) are available;

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

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

The AC power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.1.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

BASES

ACTIONS (continued)

A.1

An offsite circuit would be considered inoperable if it were not available to one required ESF train. Although two trains are required by LCO 3.8.10, the one train with offsite power available may be capable of supporting sufficient required features to allow continuation of recently irradiated fuel movement. By the allowance of the option to declare required features inoperable, with no offsite power available, appropriate restrictions will be implemented in accordance with the affected required features LCO's ACTIONS.

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3

With the offsite circuit not available to all required trains, the option would still exist to declare all required features inoperable. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability or the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC source and to continue this action until restoration is accomplished in order to provide the necessary AC power to the unit safety systems.

BASES

ACTIONS

A.2.1, A.2.2, A.2.3, B.1, B.2, and B.3 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power source should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System's ACTIONS would not be entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A are modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, the ACTIONS for LCO 3.8.10 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit, whether or not a train is de-energized. LCO 3.8.10 would provide the appropriate restrictions for the situation involving a de-energized train.

SURVEILLANCE REQUIREMENTS

SR 3.8.2.1

SR 3.8.2.1 requires the SRs from LCO 3.8.1 that are necessary for ensuring the OPERABILITY of the AC sources in other than MODES 1, 2, 3, and 4. SR 3.8.1.9 is not required to be met since only one offsite circuit is required to be OPERABLE. SR 3.8.1.16 is not required to be met because the ESF actuation signal is not required to be OPERABLE.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DG(s) from being paralleled with the offsite power network or otherwise rendered inoperable during performance of SRs, and to preclude deenergizing a required 4160 V ESF bus or disconnecting a required offsite circuit during performance of SRs. With limited AC sources available, a single event could compromise both the required circuit and the DG. It is the intent that these SRs must still be capable of being met, but actual performance is not required during periods when the DG and offsite circuit is required to be OPERABLE. Refer to the corresponding Bases for LCO 3.8.1 for a discussion of each SR.

REFERENCES

None.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources - Shutdown

BASES

BACKGROUND	A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 14 (Ref. 1), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 5 and 6 and during movement of recently irradiated fuel assemblies ensures that:

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

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

c. Adequate DC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident involving handling of recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The DC Sources - Shutdown satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

One DC electrical power subsystem, consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling within the subsystem, is required to be OPERABLE to support one subsystem of the distribution systems required OPERABLE by LCO 3.8.10, "Distribution Systems - Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents involving handling of recently irradiated fuel and inadvertent dilution events).

APPLICABILITY

The DC electrical power source required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provides assurance that:

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

b. Required features needed to mitigate a fuel handling accident involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) are available;

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

BASES

APPLICABILITY (continued)

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

The DC electrical power requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.4.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2, and A.3

With the required DC electrical power subsystem inoperable, the minimum required DC electrical power subsystem is not available. Therefore, suspension of the movement of recently irradiated fuel assemblies, and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6) is required. Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the Reactor Coolant System (RCS) for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystem and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the unit safety systems.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystem should be completed as quickly as possible in order to minimize the time during which the unit safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.3. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring the OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during the performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Inverters - Shutdown

BASES

BACKGROUND	A description of the inverters is provided in the Bases for LCO 3.8.7, "Inverters - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the USAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature systems are OPERABLE. The DC to AC inverters are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System (RCS), and containment design limits are not exceeded.

The OPERABILITY of the inverters is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

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

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

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

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident involving handling of recently irradiated fuel. Due to radioactive decay, DC electrical power is only required to mitigate fuel handling accidents involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours).

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The shutdown Technical Specification requirements are designed to ensure that the unit has the capability to mitigate the consequences of certain postulated accidents. Worst case DBAs which are analyzed for operating MODES are generally viewed not to be a significant concern during shutdown MODES due to the lower energies involved. The Technical Specifications therefore require a lesser complement of electrical equipment to be available during shutdown than is required during operating MODES. More recent work completed on the potential risks associated with shutdown, however, have found significant risk associated with certain shutdown evolutions. As a result, in addition to the requirements established in the Technical Specifications, the industry has adopted NUMARC 91-06, "Guidelines for Industry Actions to Assess Shutdown Management," as an Industry initiative to manage shutdown tasks and associated electrical support to maintain risk at an acceptable low level. This may require the availability of additional equipment beyond that required by the shutdown Technical Specifications.

The inverters were previously identified as part of the Electrical Power Distribution System and, as such, satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The inverters ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. The required inverter provides uninterruptible supply of AC electrical power to the AC instrument bus even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverter requires the associated 120 VAC instrument bus to be powered by the inverter with output voltage and frequency within tolerances, and power input to the inverter from a 125 VDC station battery. Power to an instrument bus is provided in the following order: 1) filtered AC through the inverter (referred to as "normal"); 2) DC changed to AC via the inverter (referred to as "standby"); and 3) non-filtered AC through the inverter via a static switch (referred to as "alternate"). Alternatively, power supply may be from an internal AC source via rectifier as long as the station battery is available as the uninterruptible power supply. This ensures the availability of sufficient inverter power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (i.e., fuel handling accidents involving handling of recently irradiated fuel and inadvertent dilution events).

BASES

APPLICABILITY

The inverter required to be OPERABLE in MODES 5 and 6 and during movement of recently irradiated fuel assemblies provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident are involving handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) available;

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

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

Inverter requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2, and A.3

With the required inverter inoperable, suspension of movement of recently irradiated fuel assemblies and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) specified in LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," or boron concentration (MODE 6) specified in LCO 3.9.1, "Boron Concentration," is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive Moderator Temperature Coefficient (MTC) must also be evaluated to ensure they do not result in a loss of required SDM.

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required inverter and to continue this action until restoration is accomplished in order to provide the necessary inverter power to the unit safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required inverters should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a constant voltage source transformer.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverter is functioning properly with all required circuit breakers closed and AC instrument bus energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC instrument bus. The 7 day Frequency takes into account the other indications available in the control room that alert the operator to inverter malfunctions.

REFERENCES

1. USAR, Chapter 14.

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.10 Distribution Systems - Shutdown

BASES

BACKGROUND

A description of the AC, DC, and AC instrument bus electrical power distribution systems is provided in the Bases for LCO 3.8.9, "Distribution Systems - Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident and transient analyses in the USAR, Chapter 14 (Ref. 1), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC, DC, and AC instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC instrument bus electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC, DC, and AC instrument bus electrical power distribution subsystems during MODES 5 and 6, and during movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) ensures that:

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

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

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

The distribution systems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support required features. This LCO explicitly requires energization of the portions of the electrical power distribution system necessary to support OPERABILITY of required systems, equipment, and components - all specifically addressed in each LCO and implicitly required via the definition of OPERABILITY.

BASES

LCO (continued)

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the unit in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents involving handling of recently irradiated fuel).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 5 and 6, and during movement of recently irradiated fuel assemblies, provide assurance that:

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

b. Systems needed to mitigate a fuel handling accident involving handling of recently irradiated fuel are available;

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

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

The AC, DC, and AC instrument bus electrical power distribution subsystems requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.9.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 5 or 6, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Entering LCO 3.0.3, while in MODE 1, 2, 3, or 4 would require the unit to be shutdown unnecessarily.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Although redundant required features may require redundant trains of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem train may be capable of supporting sufficient required features to allow continuation of recently irradiated fuel movement. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected

BASES

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend movement of recently irradiated fuel assemblies and operations involving positive reactivity additions that could result in loss of required SDM (MODE 5) or boron concentration (MODE 6). Suspending positive reactivity additions that could result in failure to meet the minimum SDM or boron concentration limit is required to assure continued safe operation. Introduction of coolant inventory must be from sources that have a boron concentration greater than what would be required in the RCS for minimum SDM or refueling boron concentration. This may result in an overall reduction in RCS boron concentration, but provides acceptable margin to maintaining subcritical operation. Introduction of temperature changes including temperature increases when operating with a positive MTC must also be evaluated to ensure they do not result in a loss of required SDM.

Suspension of these activities does not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the unit safety systems.

Notwithstanding performance of the above conservative Required Actions, a required residual heat removal (RHR) subsystem may be inoperable. In this case, Required Actions A.2.1 through A.2.3 do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR ACTIONS would not be entered. Therefore, Required Action A.2.4 is provided to direct declaring RHR inoperable, which results in taking the appropriate RHR actions.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power.

BASES

SURVEILLANCE REQUIREMENTS	SR 3.8.10.1 This Surveillance verifies that the AC, DC, and AC instrument bus electrical power distribution subsystems are functioning properly, with all the required buses energized. The verification of proper voltage availability on the required buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these required buses. The 7 day Frequency takes into account the capability of the electrical power distribution subsystems, and other indications available in the control room that alert the operator to subsystem malfunctions.
REFERENCES	1. USAR, Chapter 14.

B 3.9 REFUELING OPERATIONS

B 3.9.6 Containment Penetrations

BASES

BACKGROUND

During movement of recently irradiated fuel assemblies (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, 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 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 there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

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 well within the guidance of Regulatory Guide 1.183 (Ref. 1). Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During movement of recently irradiated fuel assemblies within containment, the equipment hatch may remain open, but must be capable of being closed ~~must be held in place by at least four bolts. Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced~~.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During movement of recently irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain capable of being closed.

BASES

BACKGROUND (continued)

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

Two systems can be used to purge or ventilate the containment; the Containment Purge and Vent System and the Post LOCA Hydrogen Control System. The Containment Purge and Vent System includes a 36 inch purge penetration and a 36 inch vent penetration. The Post LOCA Hydrogen Control System includes a 2 inch purge penetration and a 2 inch vent penetration. During MODES 1, 2, 3, and 4, the two valves in each of the normal purge and vent penetrations are secured in the closed position. The post LOCA hydrogen control subsystem contains two trains. The valves in Train A are normally closed. The valves in Train B are also normally closed but are periodically opened to control containment pressure within the required limits. The Train B valves receive a signal to close via the Engineered Safety Features Actuation System and the Containment Purge and Vent Isolation System. Neither of the systems are subject to a Specification in MODE 5.

In MODE 6, fresh, tempered air is provided to conduct refueling operations. The normal 36 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during recently irradiated fuel movements.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident involving handling of recently irradiated fuel. The fuel handling accident is a postulated event that involves damage to irradiated fuel. Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly vertically onto a rigid surface or onto other irradiated fuel assemblies. The requirements of LCO 3.9.5, "Refueling Cavity Water Level," in conjunction with a minimum decay time of 100 hours prior to irradiated fuel movement, ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in Regulatory Guide 1.183 (Ref. 1).

BASES

LCO

This LCO limits the consequences of a fuel handling accident involving handling recently irradiated fuel in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except when appropriate administrative controls are in place which ensure the capability to close the penetration ~~for the OPERABLE containment purge and vent penetrations and the containment personnel air locks~~. For the OPERABLE containment purge and vent penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Vent Isolation System.

The LCO is modified by a Note allowing penetration flow paths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. Administrative controls ensure that 1) appropriate personnel are aware of the open status of the penetration flow path during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, and 2) specified individuals are designated and readily available to isolate the flow path in the event of a fuel handling accident.

The containment personnel air lock doors may be open during movement of recently irradiated fuel in the containment provided that one door is capable of being closed within 30 minutes in the event of a fuel handling accident within containment. When both personnel airlock doors are open during the movement of irradiated fuel in the containment, appropriate plant personnel shall be notified of this condition. A specified individual(s) is designated and available to close the airlock following a required evacuation of containment. Any obstruction(s) (e.g., cables and hoses) that can prevent closure of an open airlock shall be able to be removed in a timely manner (i.e., within the 30 minutes specified above). Should a fuel handling accident occur inside containment, one personnel air lock door will be closed following an evacuation of containment.

The containment equipment hatch may be open during movement of recently irradiated fuel in the containment provided that it is capable of being closed within 45 minutes in the event of a fuel handling accident within containment. When the equipment hatch is open during the movement of irradiated fuel in the containment, appropriate plant personnel shall be notified of this condition. A specified individual(s) is designated and available to close the equipment hatch following a required evacuation of containment. Any obstruction(s) (e.g., cables and hoses) that can prevent closure of the equipment hatch within 45 minutes shall be able to be removed in a timely manner. Should a fuel handling accident occur inside containment, the equipment hatch will be closed following an evacuation of containment.

If it is determined that closure of the equipment hatch and/or containment penetrations would represent a significant radiological hazard to the personnel involved, the decision may be made to forgo closure of the hatch and/or penetrations.

APPLICABILITY

The containment penetration requirements are applicable during movement of recently irradiated fuel assemblies within containment because this is when there is a potential for the limiting fuel handling accident within containment. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when movement of irradiated fuel assemblies within containment is not being conducted, the potential for a fuel handling accident does not exist. Additionally, due to radioactive decay, a fuel handling accident involving handling recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 375 hours) will result in doses that are within the guideline values specified in 10 CFR 50.67, even without containment closure capability. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1

If the containment equipment hatch, air locks, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Purge and Vent Isolation System not capable of automatic actuation when the purge and vent valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending movement of recently irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1

This Surveillance demonstrates that each required containment penetration is in the required status. The Surveillance on the open purge and vent valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and vent isolation signal.

The Surveillance is performed every 7 days during movement of recently irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. As such, this Surveillance ensures that a postulated fuel handling accident involving handling recently irradiated fuel that releases fission product radioactivity within the containment will not result in a release of significant fission product radioactivity to the environment in excess of those recommended by Regulatory Guide 1.183 (Reference 1).

SR 3.9.6.2

This Surveillance demonstrates that each required containment purge and vent valve actuates to its isolation position on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. LCO 3.3.6, "Containment Purge and Vent Isolation Instrumentation," provides additional Surveillance Requirements for the containment purge and vent valve actuation circuitry. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident involving handling of recently irradiated fuel to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this Surveillance is not required to be met for valves in isolated penetrations. The LCO provides the option to close penetrations in lieu of requiring automatic actuation capability.

REFERENCES

1. Regulatory Guide 1.183, July 2000.
2. USAR, Section 14.2.1.

ATTACHMENT 4

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

RADIOLOGICAL ACCIDENT ANALYSIS AND DISCUSSION OF ASSOCIATED TECHNICAL SPECIFICATION CHANGES

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

Radiological Accident Analyses
and
Discussion of Associated Technical Specification Changes

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1.0 Introduction & Background

1.1 Introduction

This report describes the evaluations conducted to assess off-site doses and control room habitability at Kewaunee Power Station (KPS) following postulated design basis accidents per Regulatory Guide 1.183 (Reference 1). The accident source term discussed in Reference 1 is herein referred to as the Alternative Source Term (AST).

The evaluations documented herein have employed the detailed methodology contained in RG 1.183 for use in design basis accident analyses for the AST. The results have been compared with the acceptance criteria contained either in 10 CFR 50.67 (Reference 2) or the supplemental guidance in RG 1.183.

This application, if granted, would:

- Implement revised meteorological X/Q estimates (atmospheric dispersion factors) for both off-site and control room receptors from postulated accident release points
- Revise the methodology used to analyze design basis dose consequences to include the RADTRAD-NAI code
- Decrease Reactor Coolant Specific Activity Limit and Iodine Spike in TS 3.4.16
- Decrease SG Secondary Side Activity Limit in TS 3.7.16
- Require control room isolation prior to movement of recently irradiated fuel
- Allow the containment penetrations to be open (including the equipment hatch) while moving recently irradiated fuel during refueling outages
- Revise the TS 1.1 definition of Dose Equivalent Iodine I-131 to reference Federal Guidance Report No. 11 (FGR 11)
- Require Operator action to isolate the control room within 1 hr following a Locked Rotor accident
- Require Operator action to place the control room in filtered recirculation mode within 20 minutes following a Fuel Handling Accident while moving recently irradiated fuel

The revised radiological dose analyses were performed with a controlled version of the computer code RADTRAD-NAI 1.1a (QA) (Reference 3). The RADTRAD computer code calculates the control room and offsite doses resulting from releases of radioactive isotopes based on user supplied atmospheric dispersion factors, breathing rates, occupancy factors and dose conversion factors. Innovative Technology Solutions of Albuquerque, New Mexico developed the RADTRAD code for the NRC. The original version of the NRC RADTRAD code was documented in NUREG/CR-6604 [Reference 4]. The Numerical Applications, Inc. (NAI) version of RADTRAD was originally derived from NRC/ITS RADTRAD, version 3.01. Subsequently, RADTRAD-NAI was changed to conform to NRC/ITS RADTRAD, Version 3.02 with additional modifications to improve usability. The RADTRAD-NAI code is maintained under NAI's QA program, which conforms to the requirements of 10 CFR 50, Appendix B.

Control Room Atmospheric Dispersion Factors were evaluated using the ARCON96 computer code (Reference 5), following the guidance of Regulatory Guide 1.194 (Reference 6). Evaluation of off-site Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) Atmospheric Dispersion Factors was performed with a controlled version of the Dominion computer code PAVAND (Reference 7) which is a Dominion variant of the NRC PAVAN code. The EAB and LPZ X/Q values were developed following the guidance of Regulatory Guide 1.145 (Reference 8).

1.2 Current Licensing Basis Summary

The current design basis radiological analyses that appear in the KPS Updated Safety Analysis Report (USAR) consist of assessments of the following events:

1. Loss of Coolant Accident
2. Fuel Handling Accident
3. Steam Generator Tube Rupture
4. Main Steam Line Break
5. Locked Rotor Accident

6. Rod Control Cluster Assembly (RCCA) Ejection Accident
7. Waste Gas Decay Tank Failure
8. Volume Control Tank Rupture (Atmospheric Release)

The analyses of record for the above events were previously docketed in Kewaunee Power Station (KPS) Amendment No. 166, issued March 17, 2003 (Reference 10), which implemented the AST; and Amendment No. 172, issued February 27, 2004 (Reference 11), which implemented a stretch power uprate to 1772 megawatt thermal (MWt). These approved radiological accident analyses used the analytical methods and assumptions outlined in RG 1.183. By letter dated January 30, 2006 (Reference 12), as supplemented by letter dated January 23, 2007 (Reference 13), DEK requested an amendment to modify the radiological accident analyses and associated TS. This amendment incorporated TS changes to compensate for the higher control room emergency zone (CREZ) unfiltered in-leakage measured during the American Society for Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004. The NRC approved this proposed amendment as KPS License Amendment 190 on March 8, 2007 (Reference 14).

1.3 Analysis Assumptions & Key Parameter Values

1.3.1 Selection of Events Requiring Reanalysis

Kewaunee Power Station has received approval for full implementation of the AST (as defined in Section 1.2.1 of Reference 1).

To support the licensing basis and plant operation changes discussed in Section 2.0 of this application, the following accidents were reanalyzed employing the guidance of RG 1.183:

- Loss of Coolant Accident (LOCA),
- Fuel Handling Accident (FHA),
- Steam Generator Tube Rupture (SGTR) Accident,

- Main Steam Line Break (MSLB) Accident,
- Locked Rotor Accident (LRA) and
- Rod Control Cluster Assembly (RCCA) Ejection Accident (REA).

The Waste Gas Decay Tank (WGDT) failure and Volume Control Tank (VCT) rupture (Atmospheric Release) radiological analyses are also being updated to reflect revised X/Q values determined in Section 3.1 of this application. Both analyses demonstrate acceptable dose to control room operators without credit of control room emergency ventilation or isolation as well as acceptable results to the EAB under Branch Technical Position (BTP) ETSB 11-5, Rev 0 (Reference 19).

The proposed licensing basis and plant operational changes are discussed in Section 2.0. These changes require appropriate changes to the KPS Technical Specifications, which are also described in Section 2.0 of this report. The key changes considered are listed below:

a. Revise definition of Dose Equivalent I-131 in Section 1.1 of the Technical Specifications to reference Federal Guidance Report No. 11 (Reference 15) as the source of thyroid committed dose equivalent (CDE) dose conversion factors.
b. Revise Technical Specification 3.4.16, to decrease the RCS activity limits to 0.1 μCi/gm DE I-131 and 16.4 μCi/gm DE Xe-133.
c. Revise Technical Specification 3.4.16, to decrease the pre-existing iodine spike limit from 20 μCi/gm DE I-131 to 10 μCi/gm DE I-131.
d. Revise Technical Specification 3.7.16, to decrease the SG bulk liquid concentration limit from 0.1 μCi/gm to 0.05 μCi/gm DE I-131.
e. Revise Technical Specification 3.7.10, to require isolation of the control room prior to movement of recently irradiated fuel.
f. Revise Technical Specification 3.9.6, to allow ANY containment penetrations to be open under Administrative Control (including the equipment hatch) during Refueling Operations.

g. Revise 3.3.7 to remove Actions and Surveillance Requirements associated with R23 instrumentation.
h. Revise the appropriate TS Bases Sections to reflect the above listed changes in accordance with the KPS Bases Control Program as described in Section 5.5.12 of the Technical Specifications.

It can be concluded from the discussion above that implementing the revised X/Q values, in conjunction with the proposed plant operational changes, will require reanalysis of the LOCA, FHA, SGTR, MSLB, LRA, REA, WGDT and VCT. Sections 3.2 through 3.9, respectively, provide detailed descriptions of the re-analyses for these events.

1.3.2 Analysis Assumptions & Key Parameter Values

This section describes the general analysis approach and presents analysis assumptions and key parameter values that are common to all the accident analyses.

The dose analyses documented in this application employ the Total Effective Dose Equivalent (TEDE) calculation method, as specified in RG-1.183 for AST applications. The Total Effective Dose Equivalent (TEDE) is determined at the Exclusion Area Boundary (EAB) for the worst 2-hour interval. TEDE for individuals at the Low Population Zone (LPZ) and for the KPS Control Room personnel are calculated for the assumed 30-day duration of the event.

The TEDE concept is defined to be the Deep Dose Equivalent, DDE, (from external exposure) plus the Committed Effective Dose Equivalent, CEDE, (from internal exposure). In this manner, TEDE assesses the impact of all relevant nuclides upon all body organs, in contrast with the previous single, critical organ (thyroid) concept for assessing internal exposure. CEDE dose conversion factors were taken from Table 2.1 of Federal Guidance Report 11 (Reference 15) per Section 4.1.2 of Regulatory Guide 1.183. The DDE is nominally equivalent to the Effective Dose Equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE is used in lieu of DDE in determining the contribution of external dose to the TEDE. EDE dose conversion factors were taken from Table III.1 of Federal Guidance Report 12 (Reference 16) per Section 4.1.4 of Regulatory Guide 1.183.

There are a number of analysis assumptions and plant features that are used in the analysis of all of the events. These assumptions and features are presented in Tables 1.3-1 through 1.3-5.

Table 1.3-1 Control Room Common Assumptions & Key Parameters

Assumption / Parameter	Value
Control Room Effective Volume	127,600 ft^3
Control Room Intake Flow Rate prior to Isolation	2750 cfm
Unfiltered Control Room Inleakage	800 cfm
Emergency Ventilation System Recirculation Flow Rate	2500 cfm $\pm$ 10%
Response Time for Control Room to Isolate upon Receipt of a Safety Injection (SI) Signal	10 seconds
Delay to Control Room Post Accident Recirculation Mode (CRPARS) operation following Receipt of a SI Signal • 10 sec Delay to Diesel Start-up • 63 sec Delay to Sequence Diesel to CRPARS • 60 sec Delay to Open Recirc damper	133 seconds
Control Room Filter Efficiencies	Elemental: 90% Organic: 90% Particulate: 99%
Control Room Wall Thickness:	$\geq$1.5 feet Concrete
Control Room Ceiling Thickness:	$\geq$1.5 feet Concrete
Control Room Occupancy Factors 0 – 24 hours 24 – 96 hours 96 – 720 hours	 1.0 0.6 0.4

Table 1.3-2 NSS Common Assumptions & Key Parameters

Assumption / Parameter	Value
Internal Reactor Containment Vessel Free Volume	1.32E6 ft^3
Shield Building Free Volume	3.74E5 ft^3
Shield Building Wall Thickness:	2.5 ft Concrete
Shield Building Dome Thickness:	2.0 ft Concrete
Internal Containment Inner Radius:	52.5 ft
Shield Building Inner Radius	57.5 ft

Table 1.3-3 Offsite Atmospheric Dispersion Factors (sec/m3)

Location / Duration	X/Q (sec/m^3)
Exclusion Area Boundary (EAB=1200 m radius)	
All Release Points	
0 – 2 hours	1.76E-04
Low Population Zone (LPZ=3 mile*)	
All Release Points	
0 – 8 hours	3.36E-05
8 – 24 hours	2.37E-05
1 – 4 days	1.12E-05
4 – 30 days	3.94E-06

* Conservatively calculated at 2 miles

Table 1.3-4 Control Room Atmospheric Dispersion Factors

Source / Accident / Duration	Control Room Intake X/Q (sec/m^3)	Isolated CR Worst $^{\Delta}$ In-leakage X/Q (sec/m^3)
Reactor Building Stack Exhaust (LOCA, REA & FHA)		
0 – 2 hour	4.88E-03	3.97E-03
2 – 8 hour	3.51E-03	2.95E-03
8 – 24 hour	1.37E-03	1.11E-03
24 – 96 hour	1.12E-03	8.89E-04
96 – 720 hour	9.41E-04	7.87E-04
Containment / Shield Building (LOCA & REA)		
0 – 2 hour	1.84E-03	1.74E-03
2 – 8 hour	1.23E-03	1.16E-03
8 – 24 hour	5.03E-04	4.70E-04
24 – 96 hour	4.22E-04	4.02E-04
96 – 720 hour	3.50E-04	3.28E-04
Auxiliary Building Stack Exhaust (LOCA, REA, FHA, MSLB, WGDT & VCT)		
0 – 2 hour	3.67E-03	2.90E-03
2 – 8 hour	2.83E-03	2.26E-03
8 – 24 hour	1.11E-03	8.79E-04
24 – 96 hour	7.34E-04	5.80E-04
96 – 720 hour	5.64E-04	4.47E-04
Containment Equipment Hatch (FHA)		
0 – 2 hour	3.41E-03	4.58E-03
2 – 8 hour	2.88E-03	3.88E-03
8 – 24 hour	1.22E-03	1.64E-03
24 – 96 hour	9.71E-04	1.32E-03
96 – 720 hour	7.66E-04	1.07E-03
Fuel Area Roll-up Door (FHA)		
0 – 2 hour	1.44E-03	1.53E-03
2 – 8 hour	1.26E-03	1.35E-03
8 – 24 hour	5.27E-04	5.61E-04
24 – 96 hour	4.23E-04	4.51E-04
96 – 720 hour	3.56E-04	3.83E-04

Table 1.3-4 Control Room Atmospheric Dispersion Factors

Source / Accident / Duration	Control Room Intake X/Q (sec/m^3)	Isolated CR Worst $^{\Delta}$ In-leakage X/Q (sec/m^3)
"A" Steam Generator PORV (MSLB, SGTR, LRA & REA)		
0 – 2 hour	2.24E-03	2.46E-03
2 – 8 hour	1.90E-03	2.13E-03
8 – 24 hour	7.69E-04	8.60E-04
24 – 96 hour	6.37E-04	6.96E-04
96 – 720 hour	5.19E-04	5.81E-04
"A" Steam Generator Safeties	Not Used Bounded by "A" SG PORV	Not Used Bounded by "A" SG PORV
"A" Steam Generator Dumps	Not Used Bounded by "A" SG PORV	Not Used Bounded by "A" SG PORV
"B" Steam Generator PORV (MSLB, SGTR, LRA & REA)		
0 – 2 hour	3.96E-02*	2.92E-02*
2 – 8 hour	3.20E-02*	2.34E-02*
8 – 24 hour	1.21E-02	8.67E-03
24 – 96 hour	1.01E-02	6.97E-03
96 – 720 hour	8.58E-03	6.41E-03
"B" Steam Generator Safeties	Not Used Bounded by "B" SG PORV	Not Used Bounded by "B" SG PORV
"B" Steam Generator Dumps	Not Used Bounded by "B" SG PORV	Not Used Bounded by "B" SG PORV

Δ The most significant pathway of inleakage to the Control Room is through doorway penetrations in communication with the Turbine Building. The worst in-leakage X/Q is the highest X/Q from the following possible intake points to the Turbine Building: TB Fan Room West Louvers, TB Fan Room East Louvers, and TB Roll-up Door.

* The value displayed can be and was divided by 5 for use in the SGTR and/or LRA dose analyses. This reduction by a factor of 5 was permitted due to the steam exhaust vertical velocity exceeding the 95th percentile wind speed at the release elevations for the SGTR and LRA. Division by 5 is only applicable for the 0-2 hour interval for the SGTR and the 0-2 hour and 2-8 hour intervals for the LRA and SGTR. Justification for this reduction by a factor of 5 is given in Section 3.4.5.3 (SGTR) and Section 3.6.5.3 (LRA) and the results are shown in Tables 3.4-4 (SGTR) and 3.6-1 (LRA).

Table 1.3-5 Breathing Rates

Location / Duration		(m^3/sec)
Offsite (EAB & LPZ)		
	0 – 8 hour	3.5E-04
	8 – 24 hour	1.8E-04
	24 – 720 hour	2.3E-04
Control Room		
	0 – 720 hour	3.5E-04

2.0 Proposed Licensing Basis Changes

This section provides a summary description of the key proposed licensing basis changes that are justified with the revised KPS AST analyses contained within this attachment.

2.1 Revised Meteorological X/Q Values for Off-site and Control Room Receptors

This analysis supports a request to revise the design basis accident atmospheric dispersion factor (X/Q) values for KPS. Atmospheric dispersion factors are significant inputs in assessments performed to demonstrate compliance with 10 CFR Part 50. The determinations of off-site and control room X/Q values were made pursuant to the guidance of Regulatory Guides 1.145 and 1.194, respectively. After approval of this licensing basis change, the X/Q used in evaluating the consequences of design basis accidents will become the official and documented values in the USAR.

2.2 Methodology Used to Analyze Dose Consequences Using the RADTRAD-NAI Code

This analysis supports a request to revise the methodology used to evaluate design basis accident dose consequences to include using the RADTRAD-NAI code. Currently approved analyses-of-record were developed by Westinghouse using proprietary codes and methods. Dominion re-analyses using RADTRAD-NAI will replace the existing Westinghouse methodology used in evaluating the dose consequences of design basis accidents and continue to follow the guidance of RG 1.183. This license amendment application is made pursuant to the requirements of 10 CFR 50 which specifies that a revision to the methodology described in the Updated Safety Analysis Report, such as the design basis radiological consequence analyses, shall be submitted for approval. The proposed changes for radiological events have been analyzed and result in acceptable consequences, meeting the criteria as specified in 10 CFR 50.67 and RG 1.183.

2.3 Maximum Coolant Activity Limits in TS 3.4.16

The limits on maximum primary coolant activity ensure that the analyzed post-accident dose consequences of design basis accidents meet the limits specified in GDC 19 and 10 CFR 50.67. The proposed change involves decreasing the reactor coolant specific activity limits to $\leq$ 0.1 µCi/gram DE I-131 and $\leq$16.4 µCi/gram DE Xe-133. The DE Xe-133 limit is set to be consistent with the level of fuel damage equivalent to 0.1 µCi/gram DE I-131 (i.e., ~0.03% failed fuel). The pre-existing iodine spike threshold is also being reduced to $\leq$10 µCi/gram DE I-131, commensurate with the limit reduction in reactor coolant specific activity. The applicable accidents analyzed for this spike ensure control room and off-site post-accident doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

2.4 Steam Generator Secondary Side Activity Limit in TS 3.7.16

In conjunction with the proposed decrease in primary coolant activity, a lower secondary side activity limit of $\leq$ 0.05 µCi/gram DE I-131 is also proposed. The decreases in Technical Specification activity limits were necessary to result in acceptable dose consequences following a radiological event. The proposed changes in the primary coolant activity (Section 2.0.C) and the secondary side activity, coupled with the methods and assumptions specified by RG 1.183, result in estimated accident dose consequences meeting the acceptance criteria of 10 CFR 50.67 and RG 1.183.

2.5 Require control room isolation prior to movement of recently irradiated fuel in TS 3.7.10

The Technical Specification Refueling Operations Requirements define criteria necessary to result in acceptable dose consequences following a fuel handling accident. The proposed change to require control room isolation during movement of recently irradiated fuel is necessary to achieve acceptable control room occupant doses.

2.6 Allow ANY containment penetrations to be open under Administrative Control (including the equipment hatch) during Refueling Operations in TS 3.9.6

The analysis of the consequences from a Fuel Handling Accident (FHA) in either the containment or fuel storage pool, use the accident criteria specified by RG 1.183 and assume open penetrations in the containment and/or the fuel storage pool area. The analysis is modeled for the worst case release scenario (e.g., highest control room X/Q and bounding off-site X/Qs with a complete release of fuel bundle radioactivity over a 2-hour duration directly to the atmosphere). The resulting dose consequences continue to meet the acceptance criteria of 10 CFR 50.67 and RG 1.183.

Any open penetrations to the containment or fuel storage pool during movement of recently irradiated fuel will be identified and administratively controlled to ensure personnel and equipment are designated to promptly close the penetration(s). Administrative controls include:

- Appropriate personnel are aware that penetrations are open,
- A specified individual(s) is designated and available to close each penetration following a fuel handling event, and
- Any obstruction(s) (e.g., cables and hoses) that could prevent closure of any penetration can be quickly removed.

2.7 Removal of R-23 Credit for Control Room Isolation

Credit for the Control Room Ventilation Intake radiation monitor R-23, which provides control room isolation, is being removed. The R-23 system is not safety grade and consists of a single radiation monitor. In addition, the isolation signal generated by R-23 is only a partial signal that will not assure closure of all control room inlet and outlet ventilation dampers to provide complete control room isolation. Full control room isolation requires actions by the operator to close minor dampers that are not included in the isolation logic. Only the current Fuel Handling Accident (FHA) and Locked Rotor Accident (LRA) events use and credit the R-23 system for control room isolation. These analyses currently rely on the assumptions that Operations will take appropriate actions to isolate the control room if R-23 fails to perform its isolation function.

Removing credit for R-23 requires an alternative means to ensure control room isolation in the event of a FHA or LRA. The proposed new FHA requires that the control room be isolated prior to moving recently irradiated fuel, so therefore, R-23 is no longer required for that accident. As discussed in Section 3.6, a proposed operator action will be required within one hour following a LRA to isolate the control room. One hour is sufficient time for the operator to identify the accident, take necessary emergency steps in response to the accident, and direct action to isolate the control room and start the control room post accident recirculation system (CRPARS).

Since R-23 is not longer credited to perform any safety function, TS 3.3.7 will be modified to remove all TS Actions and Surveillance Requirements associated with R-23 instrumentation.

2.8 Definition of Dose Equivalent I-131

A change to the Technical Specification Definition of Dose Equivalent Iodine I-131 is proposed to reference Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989, as the source of thyroid CDE dose conversion factors (Reference 15).

2.9 Summary of Design and Licensing Basis Changes

This Section provides a comparative summary of the current design and licensing basis and the proposed changes. The summary is listed in Table 2.0-1. A detailed discussion of the changes, including the reasons for the changes, can be found in Section 3.

The existing analyses for the radiological events, as listed in Section 1.2, were performed at various times using different codes and/or hand calculations. The common element for these events is the use of a single "bounding" control room X/Q and a single set of off-site dispersion factors (X/Q) to assess resulting radiological consequences. The proposed amendment utilizes new estimates of control room and off-site (EAB and LPZ) dispersion factors using the guidance provided in Regulatory Guides 1.145 and 1.194, and supporting documents. Additionally, a comprehensive design basis validation for all inputs and assumptions used in each radiological analysis was performed. This accounts for differences in some of the parameters listed in Table 2.0-1.

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
	Alternate Source Term	
RCS Technical Specification Limits	1.0 μCi/gm DE I-131 AND 595 μCi/gm DE Xe-133	0.1 μCi/gm DE I-131 AND 16.4 μCi/gm DE Xe-133
RCS Technical Specification Gross Gamma Concentrations	USAR Table D.4-1 (based on 1% fuel defects)	Table 3.4-1 (based on fuel defects equivalent to 0.1 μCi/gm DE I-131)
RCS Technical Specification Iodine Concentrations <u>Isotope</u> I-131 I-132 I-133 I-134 I-135	1.0 μCi/gm DE I-131 Conc. <u>(μCi/gm)</u> 7.80E-01 7.93E-01 1.16E+00 1.61E-01 6.37E-01	0.1 μCi/gm DE I-131 Conc. <u>(μCi/gm)</u> 7.82E-02 7.97E-02 1.17E-01 1.62E-02 6.40E-02
Secondary Side Technical Specification Limit	≤ 0.1 μCi/gm DE I-131	≤ 0.05 μCi/gm DE I-131
Pre-accident Iodine Spike	20 μCi/gm DE I-131	10 μCi/gm DE I-131

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Iodine Appearance Rates Isotope I-131 I-132 I-133 I-134 I-135	Conc. (Ci/min) 0.301 0.788 0.519 0.319 0.377	Conc. (Ci/min) 0.030 0.079 0.052 0.032 0.038
Dose Conversion Factors	ICRP30	FGR 11 and 12
Offsite Dose	Historical X/Qs (unknown basis)	Revised X/Qs (based on RG 1.145 and PAVAND)
Offsite Breathing Rates 0 – 8 hours 8 – 24 hours 24 – 720 hours	 3.47E-04 1.75E-04 2.32E-04	 3.5E-04 1.8E-04 2.3E-04
Control Room		
Unfiltered Inleakage (cfm)	800 (based on SI signal) OR 1500 (based on R-23 signal)	800 credit for R-23 has been removed
X/Q's	Murphy & Campe	ARCON96 (listed in Table 1.3-4)
Breathing Rate	3.47E-04	3.5E-04
Loss-of Coolant Accident (Section 3.2)		
Iodine Chemical Form in the Sump (%)	100% Elemental	97% Elemental 3% Organic
Containment Sump Volume (gal)	315,000	311,000
Containment Spray Duration (hr)	0.917	0.91

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Containment spray Removal Coefficient (hr^{-1}) Elemental Particulate	 20 4.5	 15 2.8
Natural deposition (hr^{-1})	0.1	Power's Model set at the 10^{th} percentile
ECCS Iodine Airborne Evolution (%) 0-3 hour >3 hour	 10 1	 10 10
RWST Iodine Airborne Evolution	1%	DF=100
RWST Backleakage modeling		Section 3.2.5.5
Fuel Handling Accident (Section 3.3)		
Unfiltered Inleakage after control room isolation (cfm)	1500	800
Credited operator action	None	Control room is isolated prior to movement of recently irradiated fuel. Operator action to place control room in filtered recirculation mode within 20 minutes of FHA.
Control Room Configuration while Moving Recently Irradiated Fuel	Normal	Isolated prior to movement

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Steam Generator Tube Rupture Accident (Section 3.4)		
Release termination of Primary to Secondary Leakage for Intact Steam Generators (hours)	24	29
Duration of break flow and discharge from Affected Steam Generator (min)	30	55
Operator Action to close Affected SG PORV (min)	30	55
Total Break Flow (lbm)	0-30 min: 154,900	0-55 min: 282,100
Condenser as a release pathway	Credited	Not Credited
Iodine Spike	500	335
Iodine Spike duration	4 hours	8 hours
Main Steam Line Break Accident (Section 3.5)		
Safety Injection Signal (sec)	0	<3
Action to Align RHR (hr)	24	29
Release to Environment (hr) Unaffected SG Affected SG	0 – 24	0 – 29

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
Pre-accident spike Concurrent spike	72 72	69.2 8
Operator Action - close Affected SG MSIV (hr)	NA	8
Release of Initial Mass in Faulted Generator (min)	2	10
Accident-Initiated (Concurrent) Spike Duration (hr)	4	8
Duration of Primary to Secondary Leakage for Affected Steam Generator	72 hours	69.2 hours
Duration of Primary to Secondary Leakage for Intact Steam Generators	24 hours	29 hours
Locked Rotor Accident (Section 3.6)		
Failed Fuel Following the Accident (%)	50	25
Steam Generator Liquid Mass (lbm/SG) 0 – 30 minutes	87,000	84,000
Control Room Isolation (min)	10.67	60
RCCA Ejection Accident (Section 3.7)		
Safety Injection Signal (sec)	52.5	240
Steam Generator Liquid Mass (lbm/SG)	87,000	84,000

Table 2.0-1 Comparative Summary of Design and Licensing Basis Changes to Radiological Event Analyses

Parameter	Current Basis	Proposed Basis
WGDT Accident (Section 3.8)		
Dose Consequence Multiplier (Method to adjust cycle activity to account for changes in operating conditions and fuel management variations)	1.1	1.12
Release Duration (min)	5	120
Release Rate (%/day)	1.99E+05	8.289E+03
Control Room Isolation (min)	0.5	30
VCT Accident (Section 3.9)		
Dose Consequence Multiplier (Method to adjust cycle activity to account for changes in operating conditions and fuel management variations)	1.1	1.12
Control Room Unfiltered Inleakage (cfm)	0	200

3.0 Radiological Event Re-Analyses & Evaluation

As documented in Section 1.3.1, this application involves the reanalysis of the design basis radiological analyses for the following accidents:

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

The calculated radiological consequences are compared with the limits provided in 10 CFR 50.67(b)(2), and as clarified per the additional guidance in RG-1.183 for events with a higher probability of occurrence.

New atmospheric dispersion factors (X/Qs) have been calculated. New control room X/Qs were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The most limiting onsite accident X/Qs were selected from source/receptor pairs which included those potentially associated with single failure, loss of offsite power, control room pre-isolation and post-isolation intake and inleakage points. The offsite Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) X/Qs were calculated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

Dose calculations are performed at the EAB for the worst 2-hour period, and for the LPZ and KPS Control Room for the duration of the accident (30 days). All of the radiological dose consequence calculations were performed with the RADTRAD-NAI computer code system (Reference 3). The dose acceptance criteria that apply are provided in Table 3.0-1.

Table 3.0-1 Accident Dose Acceptance Criteria

Accident or Case	Control Room(1)	EAB & LPZ
Design Basis LOCA	5 rem TEDE	25 rem TEDE
Steam Generator Tube Rupture Fuel Damage or Pre-accident Spike Coincident Iodine Spike	 5 rem TEDE 5 rem TEDE	 25 rem TEDE 2.5 rem TEDE(2)
Main Steam Line Break Fuel Damage or Pre-accident Spike Coincident Iodine Spike	 5 rem TEDE 5 rem TEDE	 25 rem TEDE 2.5 rem TEDE(2)
Locked Rotor Accident	5 rem TEDE	2.5 rem TEDE(2)
RCCA Ejection Accident	5 rem TEDE	6.3 rem TEDE(2)
Fuel Handling Accident	5 rem TEDE	6.3 rem TEDE(2)
Waste Gas Decay Tank Failure	5 rem TEDE	0.5 rem WB(3)
Volume Control Tank Rupture	5 rem TEDE	0.5 rem WB(3)

(1) Based on 10CFR50.67 and 10 CFR 50, Appendix A, GDC 19
(2) Reduced from 10 CFR 50.67 criteria in accordance with RG 1.183 for higher probability events.
(3) Current licensing basis

3.1 Determination of Atmospheric Dispersion Factors (X/Q)

A comprehensive evaluation of X/Q values applicable to the radiological events listed in Section 1.3.1 has been performed. Release points for each accident scenario were identified and paired with possible receptor locations to determine the most limiting X/Q values. The most limiting X/Q values were used to model the dose consequences. Onsite source/receptor pairs were evaluated using the qualified and tested ARCON96 code (Reference 5) while the offsite source/receptor pairs to the EAB and LPZ were evaluated with a controlled version of the Dominion computer code PAVAND (Reference 7) which is a Dominion variant of the NRC PAVAN code.

Enclosure 1 of this Attachment includes a computer file on CDROM which contains the site meteorological data collected over the years 2002-2006 and used as the primary input in the calculation of the atmospheric dispersion factors . The meteorological data for KPS collected over this period were collected and processed in accordance with the standards described in RG 1.23 (Reference 18). Additionally, Enclosure 1 also includes the ARCON96 and PAVAND input files that were used in the calculation of the control room and offsite X/Q values.

The meteorological data is hourly as described in Regulatory Guide 1.23. This data has been reviewed by meteorologists for missing or anomalous observations, instrumentation problems, and trends indicative of local effects such as building wakes and excessive vegetation effects. The data meets the requirement of Regulatory Guide 1.23 for annual joint recovery rates of at least 90%.

During the review of the meteorological data, the meteorologists observed that there was a change in the distribution of the atmospheric stability classes in the data during early January of 2005. After January 2005, the occurrence of extremely and moderately unstable stability classes increased from the distribution observed from the previous three years of data. At the same time, the occurrence of slightly stable stability classes decreased. An effort was made to determine the cause of this shift in stability class distribution. During January of 2005, the Kewaunee plant process computer was replaced. The algorithm used to calculate the stability class was examined. The algorithm was found to comply with requirements and methods. The stability classes since Jan 2005 were compared to available Point Beach data and they matched well. Point Beach is located just a few miles south of KPS. The conclusion reached was that the change in stability class distribution was tied to the replacement of the plant process computer. However, no conclusion could be reached on whether the stability class distribution, before the plant process computer change, was necessarily incorrect.

Intuitively, an increase in the percentage of highly unstable wind conditions should cause the resulting atmospheric dispersion factors to be smaller. Based on the stability class distribution, it was believed that use of only the final 2 years of data would result in smaller X/Q values. Use of only the first 3 years of data could be overly conservative. Since the last two years of data meet quality standards and compare favorably to data recorded for the same period at Point Beach, the use of only the first 3 years of data, which contain a larger distribution of stable atmospheric conditions for unknown reasons, did not seem appropriate. Therefore, the meteorological data for all 5 years were used and are believed to be appropriate and conservative.

3.1.1 Control Room X/Q

Control room X/Qs are calculated for both ventilation intake and potential inleakage receptor points to the control room and are listed in Table 1.3-4. Figure 3.1-1 provides a relative scaled drawing of the KPS building orientation and control room location showing all identified release points and receptors. The control room envelope is physically within the Auxiliary Building with ingress/egress doors into both the Auxiliary and Turbine Buildings.

DEK believes the primary source of inleakage into the control room occurs through the ingress/egress doors. This conclusion is based on the following:

a. In December of 2004, tracer gas tests were performed to measure the unfiltered in-leakage into the KPS control room. Based on observations and measurements obtained during those tests, the ingress/egress doors appeared to be the most viable source of inleakage when the control room is isolated.

b. The isolation dampers in the normal and alternate control room intakes are bubble-tight dampers. Due to the nature of their design, no inleakage is expected to occur past these dampers when closed.
c. Due to multiple areas within the Auxiliary Building being under suction by the Special Ventilation System, some directly adjacent to the control room boundary, the primary pathway and source of inleakage through the control room doors is considered to be from the turbine building.

Due to the facts above, the most viable intake to the control room when the normal control room intake is isolated is from the Turbine Building through the ingress/egress doors. Various intake points to the Turbine Building were considered as receptor locations and are shown in Figure 3.1-1. These locations are: Turbine Building Fan Room West Louver, Turbine Building Fan Room East Louver, and the Turbine Building Roll-up Door. No credit is taken for dilution within the large Turbine Building volume or additional dispersion within the Turbine Building as the contaminants travel from the intake point to the likely control room inleakage doorways. In essence, the intake into the Turbine Building is being conservatively treated very similar to a ventilation duct leading directly to the control room.

As a result of the analyses documented in this LAR, the alternate control room intake will be restricted from use. This restriction is required because of the X/Q that would result due to the close proximity of the alternate intake to various release points; one of which is < 10 m from the alternate intake. Administrative controls will be in place to assure the alternate control room intake is closed and prohibit its use during normal operation, following an accident, or while moving recently irradiated fuel.

Control room X/Q values for the source/receptor pairs address the most viable locations and limiting accident cases, including those potentially associated with single failure and loss of offsite power. The ARCON96 input source-to-receptor distances were the shortest horizontal (X-Y) distance between the release point and intake, regardless of intervening buildings (i.e., source to receptor taunt-strings or elevation differences were not considered). Table 3.1-1 and 3.1-2 provide the distances and angles for each source to receptor combination.

In accordance with the guidance of RG 1.194, the buoyant plume rise associated with energetic releases from steam relief values or atmospheric steam dumps can be credited if (1) the release is uncapped and vertical, and (2) the time-dependent vertical velocity exceeds the 95^{th} percentile wind speed, at the release point height, by a factor of 5. Justification for crediting buoyant plume rise is given in Section 3.4.5.3 (SGTR) and Section 3.6.5.3 (LRA).

Figure 3.1-1 Kewaunee Source and Receptor Points

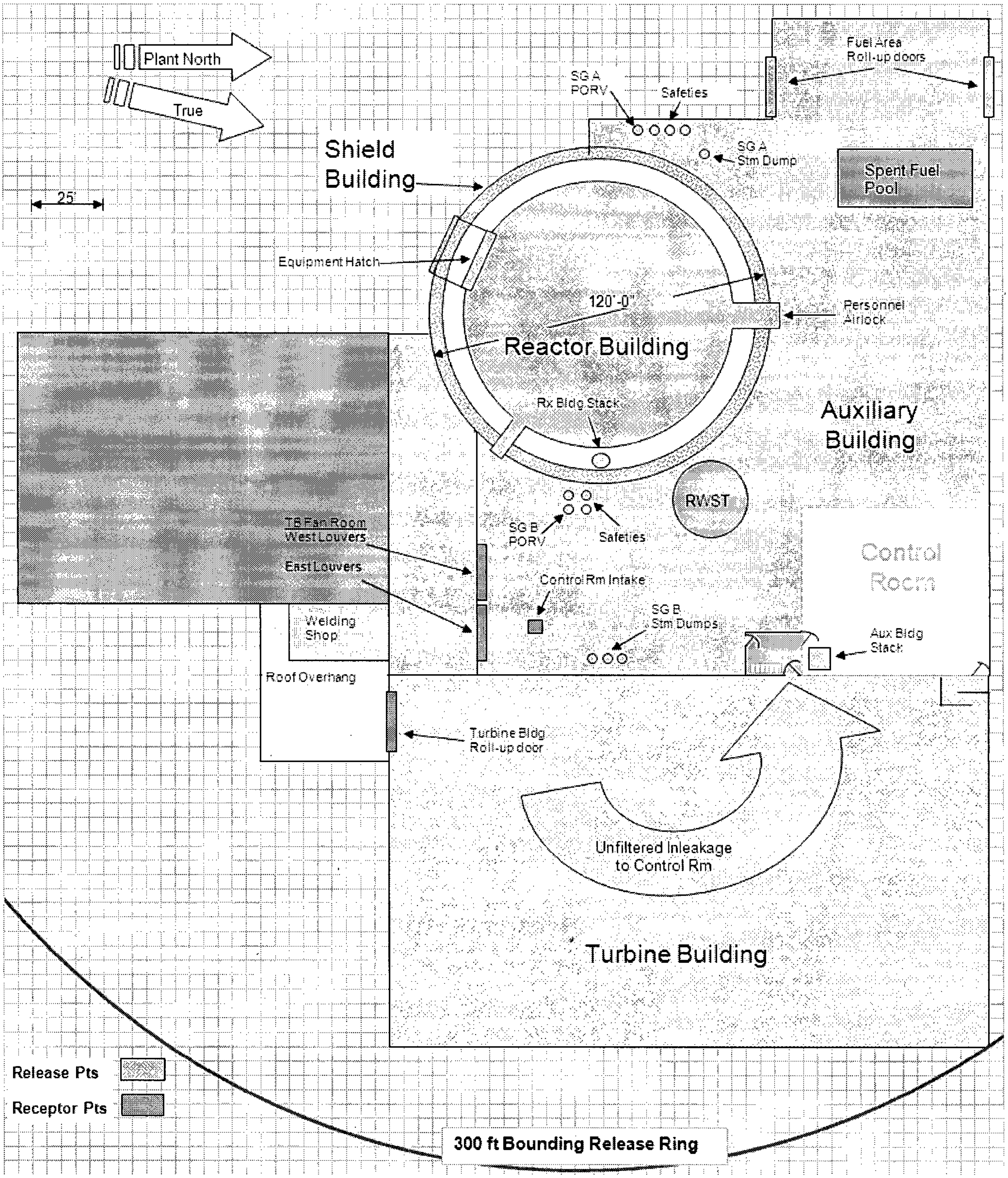


Table 3.1-1 Line-of-Sight Horizontal Distance from Source to Receptor

(meters)

		RECEPTORS			
*Elev.**		*21.69 m*	*17.91m*	*17.91 m*	*3.51 m*
	RELEASE POINTS	*Control Room Intake*	*TB Fan Room West Louver*	*TB Fan Room East Louver*	*TB Roll-up Door*
51.35 m	Rx Bldg Stack	17.05	16.93	21.67	33.61
37.13 m	Shield Bldg	14.58	12.34	18.45	29.83
27.89 m	Aux Bldg Stack	39.60	44.89	44.20	53.23
4.27 m	Equipment Hatch	39.60	34.61	41.46	50.39
3.51 m	Fuel Area Roll-up Doors[1]	64.55	62.98	68.83	80.44
12.60 m	SG A PORV	53.35	50.87	57.14	68.33
22.35 m	SG A Dump	57.93	56.53	62.27	73.95
12.60 m	SG A Safeties[2]	53.79	51.67	57.83	69.15
23.34 m	SG B PORV	12.06	12.81	16.84	28.82
25.83 m	SG B Dumps[3]	24.81	30.56	29.16	38.00
23.34 m	SG B Safeties[4]	13.25	13.46	17.90	29.85

* Above grade (meters)
1 Fuel Area Roll-up Door #2 (south) to all receptors
2 Safety #2 to CR Intake, TB FR East Louver; Safety #1 to TB FR West Louver, TB Roll-Up Door
3 Dump #1 (South) for all receptors
4 Safety #1 to all receptors

Table 3.1-2 Direction from Receptor to Source
(degrees true North)

RELEASE POINTS	RECEPTORS			
	Control Room Intake	*TB Fan Room West Louver*	*TB Fan Room East Louver*	*TB Roll-up Door*
Rx Bldg Stack	286.4°	308.7°	293.9°	296.1°
Shield Bldg	273.3°	284.4°	279.1°	284.2°
Aux Bldg Stack	349.4°	354.8°	346.0°	336.7°
Equipment Hatch	246.0°	252.6°	252.9°	263.0°
Fuel Area Roll-up Doors[1]	282.0°	287.8°	284.6°	286.9°
SG A PORV	273.2°	279.9°	277.0°	280.9°
SG A Dump	283.1°	289.6°	285.9°	288.2°
SG A Safeties[2]	277.4°	281.6°	280.8°	282.2°
SG B PORV	289.7°	320.2°	298.4°	299.1°
SG B Dumps[3]	356.2°	2.8°	349.9°	336.0°
SG B Safeties[4]	286.2°	314.6°	295.3°	297.2°

1 Fuel Area Roll-up Door #2 (south) to all receptors
2 Safety #2 to CR Intake, TB FR East Louver; Safety #1 to TB FR West Louver, TB Roll-Up Door
3 Dump #1 (South) for all receptors
4 Safety #1 to all receptors

3.1.2 Offsite (EAB and LPZ) X/Q

The Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors (X/Qs) for Kewaunee Power Station have been revised and are listed in Table 1.3-3. Generated using the PAVAND code, the X/Qs are based upon a conservatively modeled ring with a 300-foot radius centered on Containment. This 300 foot bounding release ring, partially shown in Figure 3.1-1, encompasses all possible release points that exist within the station and is based upon the distance from the center of Containment to the farthest release point (i.e., Northeast Turbine Building corner). All actual release points are contained within this 300 foot bounding ring. The EAB and LPZ X/Q values were conservatively modeled using a ground-level release without credit for building wake effects.

Figure 2.2-2 in the KPS USAR shows the KPS EAB as an exclusion radius of 1,200 meters. The exclusion radius over land falls within the physical site boundary. For conservatism, the LPZ was calculated assuming the bounding shortest radius of 2 miles (3218.7 m). Utilizing the 300-foot (91.4 meters) bounding release ring described above, the shortest distance to the EAB (3,637 ft or 1,108.6 m) and the LPZ (10,260 ft or 3,127.3 m) for all directions (centered on the containment) was used to represent the bounding assumption for all possible release points. Modeled as a ground level release, the resulting EAB and LPZ X/Qs were determined by selecting the largest calculated value across all sixteen downwind directions and the overall site for each prescribed time period. The EAB (0-2 hour) X/Q is a single bounding value of 1.76E-04 sec/m^3. The LPZ (0-8 hr, 8-24 hr, 1-4 day, and 4-30 day) X/Qs represent the highest calculated values for each time period across all directions. The maximum values occurred in the East-Northeast (ENE) direction for all except one time period, the (4-30 day) period, which occurred in the East (E) direction. Selecting the highest value within each time period across all directions and the overall site assures that the doses calculated for the LPZ are conservative.

3.2 Design Basis Loss of Coolant Accident (LOCA) Reanalysis

Amendment 190 (TAC No. MC9715, ADAMS Accession No. ML070430020) for Kewaunee Power Station, dated March 8, 2007 (Reference 14), represents the current licensing basis for the LOCA. This amendment incorporated TS changes to compensate for the higher control room emergency zone (CREZ) unfiltered in-leakage measured during the American Society for Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004 (Reference 20).

This section describes the methods employed and results obtained from the radiological reanalysis of the design basis LOCA. The analysis considers dose from several sources. They are:

- Containment Leakage Plume,
- Emergency Core Cooling System (ECCS) Component Leakage
- Refueling Water Storage Tank Vent
- Containment, Plume, and Filter Shine are negligible to control room occupants based on control room structure boundaries, penetration pathways and internal shield walls consisting of at least, or equivalent to, 18 inches of concrete; based on NUREG-0800, Section 6.4, "Control Room Habitability System" (Reference 21)
- Containment purge isolates within 37 seconds following the LOCA and is an insignificant contributor to control room and offsite dose.

Doses are calculated at the Exclusion Area Boundary (EAB) for the worst-case two-hour period, at the Low Population Zone Boundary (LPZ), and in the KPS Control Room. The methodology used to evaluate the doses resulting from a LOCA is consistent with RG 1.183 (Reference 1).

3.2.1 LOCA Scenario Description

The design basis LOCA scenario for radiological calculations is initiated assuming a major rupture of the primary reactor coolant system piping. In order to yield radioactive releases of the magnitude specified in RG 1.183, it is also assumed that the ECCS does not provide adequate core cooling, such that significant core melting occurs. This general scenario does not represent any specific accident sequence, but is representative of a class of severe damage incidents that were evaluated in the development of the RG 1.183 source term characteristics. Such a scenario would be expected to require multiple failures of systems and equipment and lies beyond the severity of incidents evaluated for design basis transient analysis. Activity from the core is released to the containment, and from there released to the environment by means of containment leakage and leakage from the emergency core cooling system. For the containment leakage analysis, all activity released from the fuel is assumed to be in the containment atmosphere until removed by sprays, sedimentation, radioactive decay or leakage from the containment. For the ECCS leakage analysis, all iodine activity released from the fuel is assumed to be in the sump solution until removed by radioactive decay or leakage from the ECCS.

3.2.2 LOCA Source Term Definition

RG 1.183 provides explicit description of the key AST characteristics recommended for use in design basis radiological analyses. The core radionuclide inventory used in this analysis was previously generated using the ORIGEN2 code for a Stretch Power Uprate (SPU) to 1772 megawatt thermal (MWt) and used in KPS Amendment No. 172, issued February 27, 2004 (Reference 11). Table 3.2-1 lists the RG 1.183 source term assumptions used in the LOCA analysis, which includes: the core inventory release fractions by radionuclide group, timing of release, and chemical form of the release into containment.

RG-1.183 divides the releases from the core into two phases:

1. The Fuel Gap Release Phase during the first 30 minutes and
2. The Early In-vessel Release Phase in the subsequent 1.3 hours.

Table 3.2-2 shows the fractions of the total core inventory of various isotope groups that are assumed released in each of the two phases of the LOCA analysis. Table 3.2-3 lists the isotopes and the associated curies at the end of a fuel cycle that was input to RADTRAD-NAI. The core inventory used in the LOCA analysis is the same source term used in Amendment No. 172, augmented with some additional core curies for Rb-88 and Cs-138 (Reference 11). Table 3.2-3 also provides the CEDE and EDE dose conversion factors for each of the isotopes. These dose conversion factors were taken from Federal Guidance Reports 11 and 12 (References 15 and 16, respectively).

Table 3.2-1 Regulatory Guide 1.183 Source Terms

Characteristic	RG 1.183 Source Term
Core Fractions Released To Containment	Noble Gases 100% Iodine 40% Cesium 30% Tellurium 5% Barium 2% Others – 0.02% to 0.25%
Timing of Release	Released in Two Phases Over 1.8 hour Interval
Iodine Chemical and Physical Form	4.85% Inorganic Vapor 0.15% Organic Vapor 95% Aerosol
Solids	Treated as an Aerosol

Table 3.2-2 RG 1.183 Release Phases

	Core Release Fractions[a]	
Isotope Group	**Gap**	**Early In-Vessel**
Noble Gases[b]	0.05	0.95
Halogens	0.05	0.35
Alkali Metals	0.05	0.25
Tellurium	0	0.05
Barium, Strontium	0	0.02
Noble Metals	0	0.0025
Cerium	0	0.0005
Lanthanides	0	0.0002
Duration (hours)	0.5	1.3

a. Release duration apply only to the Containment release. The ECCS leakage portion of the analysis conservatively assumes that the entire core release fraction is in the containment sump from the start of the LOCA.

b. Noble Gases are not scrubbed from the containment atmosphere and therefore are not found in either the sump or ECCS fluid.

Table 3.2-3 Core Inventory and Dose Conversion Factors by Isotope (1782.6 MWt with 1.06 Multiplier*)

Isotope	Isotope Group	Curies	EDE $Sv\text{-}m^3/Bq\text{-}sec$	CEDE Sv/Bq
Kr-85	Noble gas	5.71E+05	1.190E-16	0.000E+00
Kr-85m	Noble gas	1.39E+07	7.480E-15	0.000E+00
Kr-87	Noble gas	2.68E+07	4.120E-14	0.000E+00
Kr-88	Noble gas	3.77E+07	1.020E-13	0.000E+00
Xe-131m	Noble gas	5.64E+05	3.890E-16	0.000E+00
Xe-133	Noble gas	9.98E+07	1.560E-15	0.000E+00
Xe-133m	Noble gas	3.05E+06	1.370E-15	0.000E+00
Xe-135	Noble gas	2.77E+07	1.190E-14	0.000E+00
Xe-135m	Noble gas	2.03E+07	2.040E-14	0.000E+00
Xe-138	Noble gas	8.65E+07	5.770E-14	0.000E+00
I-131	Halogen	5.04E+07	1.820E-14	8.890E-09
I-132	Halogen	7.33E+07	1.120E-13	1.030E-10
I-133	Halogen	1.04E+08	2.940E-14	1.580E-09
I-134	Halogen	1.14E+08	1.300E-13	3.550E-11
I-135	Halogen	9.73E+07	8.294E-14	3.320E-10
Rb-86	Alkali Metal	1.11E+05	4.810E-15	1.790E-09
Rb-88	Alkali Metal	3.77E+07	3.360E-14	2.260E-11
Cs-134	Alkali Metal	9.82E+06	7.570E-14	1.250E-08
Cs-136	Alkali Metal	2.80E+06	1.060E-13	1.980E-09
Cs-137	Alkali Metal	6.09E+06	2.725E-14	8.630E-09
Cs-138	Alkali Metal	8.65E+07	1.210E-13	2.740E-11
Sb-127	Tellurium	5.36E+06	3.330E-14	1.630E-09
Sb-129	Tellurium	1.62E+07	7.140E-14	1.740E-10
Te-127	Tellurium	5.31E+06	2.420E-16	8.600E-11
Te-127m	Tellurium	6.90E+05	1.470E-16	5.810E-09
Te-129	Tellurium	1.59E+07	2.750E-15	2.090E-11
Te-129m	Tellurium	2.35E+06	3.337E-15	6.484E-09

Table 3.2-3 Core Inventory and Dose Conversion Factors by Isotope (1782.6 MWt with 1.06 Multiplier*)

Isotope	Isotope Group	Curies	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
Te-131	Tellurium	0.00E+00 ‡	2.040E-14	1.290E-10
Te-131m	Tellurium	7.31E+06	7.463E-14	1.758E-09
Te-132	Tellurium	7.21E+07	1.030E-14	2.550E-09
Sr-89	Barium-Strontium	5.11E+07	7.730E-17	1.120E-08
Sr-90	Barium-Strontium	4.51E+06	7.530E-18	3.510E-07
Sr-91	Barium-Strontium	6.33E+07	4.924E-14	4.547E-10
Sr-92	Barium-Strontium	6.82E+07	6.790E-14	2.180E-10
Ba-139	Barium-Strontium	9.34E+07	2.170E-15	4.640E-11
Ba-140	Barium-Strontium	8.99E+07	8.580E-15	1.010E-09
Mo-99	Noble Metal	9.63E+07	7.280E-15	1.070E-09
Rh-105	Noble Metal	4.71E+07	3.720E-15	2.580E-10
Ru-103	Noble Metal	7.59E+07	2.251E-14	2.421E-09
Ru-105	Noble Metal	5.10E+07	3.810E-14	1.230E-10
Ru-106	Noble Metal	2.52E+07	1.040E-14	1.290E-07
Tc-99m	Noble Metal	8.44E+07	5.890E-15	8.800E-12
Ce-141	Cerium	8.54E+07	3.430E-15	2.420E-09
Ce-143	Cerium	7.97E+07	1.290E-14	9.160E-10
Ce-144	Cerium	6.54E+07	2.773E-15	1.010E-07
Np-239	Cerium	1.01E+09	7.690E-15	6.780E-10
Pu-238	Cerium	1.90E+05	4.880E-18	7.790E-05
Pu-239	Cerium	1.93E+04	4.240E-18	8.330E-05
Pu-240	Cerium	2.67E+04	4.750E-18	8.330E-05
Pu-241	Cerium	6.24E+06	7.250E-20	1.340E-06
Am-241	Lanthanides	7.56E+03	8.180E-16	1.200E-04
Cm-242	Lanthanides	1.62E+06	5.690E-18	4.670E-06
Cm-244	Lanthanides	1.66E+05	4.910E-18	6.700E-05
La-140	Lanthanides	9.76E+07	1.170E-13	1.310E-09

Table 3.2-3 Core Inventory and Dose Conversion Factors by Isotope (1782.6 MWt with 1.06 Multiplier*)

Isotope	Isotope Group	Curies	EDE Sv-m^3/Bq-sec	CEDE Sv/Bq
La-141	Lanthanides	8.53E+07	2.390E-15	1.570E-10
La-142	Lanthanides	8.26E+07	1.440E-13	6.840E-11
Nb-95	Lanthanides	8.77E+07	3.740E-14	1.570E-09
Nd-147	Lanthanides	3.40E+07	6.190E-15	1.850E-09
Pr-143	Lanthanides	7.70E+07	2.100E-17	2.190E-09
Y-90	Lanthanides	4.68E+06	1.900E-16	2.280E-09
Y-91	Lanthanides	6.55E+07	2.600E-16	1.320E-08
Y-92	Lanthanides	6.85E+07	1.300E-14	2.110E-10
Y-93	Lanthanides	7.87E+07	4.800E-15	5.820E-10
Zr-95	Lanthanides	8.71E+07	3.600E-14	6.390E-09
Zr-97	Lanthanides	8.62E+07	4.432E-14	1.171E-09

* increased by 6% to account for variations in core parameters: 493.6 ± 10% EFPD, average enrichment of 4.5 w/o ± 10%, and core mass of 49.1 MTU ± 10%.

‡ Although Te-131 was not included in the initial core inventory, it was included in the analysis as a significant decay product.

3.2.3 LOCA Atmospheric Dispersion Factors

3.2.3.1 LOCA Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The LOCA Control Room X/Qs listed in Table 1.3-4 were calculated for the following KPS source points:

- Reactor Building Exhaust Stack
- Shield Building
- Auxiliary Building Exhaust Stack

3.2.3.2 LOCA Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.2.4 LOCA Containment Airborne Activity

3.2.4.1 Containment Sprays

The current licensing basis for the LOCA uses containment sprays to remove elemental and particulate iodine from the containment atmosphere. The use of containment sprays and methods to determine elemental and particulate iodine removal rates were approved in KPS Amendment No. 166, issued March 17, 2003 (Reference 10).

One train of the containment spray system is assumed to operate following the LOCA. Injection spray is credited with no delay in startup. Earlier spray actuation is conservative since it results in earlier spray termination. There is no benefit from earlier spray actuation since there is little activity in the containment at the time the spray starts. When the RWST drains to a predetermined setpoint level, the operators switch to recirculation of sump liquid. Switchover to recirculation spray is not credited in the analysis and all spray is assumed to be terminated when the RWST drains down. The analysis conservatively assumes that the sprays are terminated 0.91 hours after the start of the event. New spray removal rates were determined based on the revised assumptions.

KPS containment spray design consists of four spray ring headers. The elemental and particulate iodine removal rates due to sprays are listed in Table 3.2-5. These spray removal rates are used until the RWST is secured at 0.91 hours. At that time, further iodine removal is ignored due to sprays even though the recirculation spray system remains operating. An elemental iodine DF of 200 and particulate iodine DF of 50 are not achieved during the period that sprays are assumed operating. Therefore, the elemental and particulate iodine removal rates remain constant during this period..

3.2.4.1.1 Containment Spray Removal of Elemental Iodine

NUREG-0800, Section 6.5.2, Rev. 2 (Reference 22) identifies a methodology previously used and approved for the determination of spray removal of elemental iodine. The removal rate constant is determined by:

$$\lambda = (6\ K_g\ T\ F)\ /\ V\ D$$

where;

λ	= elemental iodine removal coefficient,
K_g	= Gas phase mass transfer coefficient
T	= Time of fall of the spray drops
F	= Volume flow rate of sprays
V	= Containment sprayed volume
D	= Mass-mean diameter of the spray drops

The spray parameter values are listed in Table 3.2-4.

These parameters and the appropriate conversion factors were used to calculate the elemental spray removal coefficient. The calculated value of 15 hr^{-1} is modeled for removal of elemental iodine from the containment atmosphere. The maximum DF of 200 is not achieved prior to assumed spray termination at 0.91 hours.

3.2.4.1.2 Containment Spray Removal of Particulates

The particulate removal coefficient was calculated using a Regulatory Guide 1.183 prescribed method from NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays" (Reference 23). Inputs to the methodology include the fall height

"H" of the water droplets in meters and the spray water flux "Q" in (cm^3 of H_2O)/(cm^2sec). Values for the fall height and spray water flux are given in Table 3.2-4.

The current particulate removal coefficient is calculated using the model described in NUREG-0800, Section 6.5.2, Revision 2 (Reference 22). Both the NUREG/CR-5966 and SRP 6.5.2 methods are deemed acceptable per RG 1.183. DEK has elected to change to the NUREG/CR-5966 method for KPS based solely on commonality to methods used at other Dominion facilities. A comparison of resulting particulate removal constants from both methods was made to determine if this change in methodology provides a benefit. The NUREG/CR-5966 method produces a smaller, more conservative coefficient that is used in the revised LOCA analysis.

NUREG/CR-5966 [Page 173] presents the following equations for aerosol (i.e., aerosol treated as particulate in SRP methodology) removal rate at the 10^{th} percentile level:

$$\ln\left(\lambda_{m_f=0.9}\right) = 5.5750 + (0.94362)\ln Q - (7.327\text{E}-7)QH^2 - (6.9821\text{E}-3)Q^2H + (3.555\text{E}-6)Q^2H^2$$

$$\frac{\lambda_{m_f}}{\lambda_{m_f=0.9}} = \left[0.1108 - (0.00201)\log_{10} Q\right]\left[1 - \left(\frac{m_f}{0.9}\right)^{0.8945}\right] + \left(\frac{m_f}{0.9}\right)^{0.8945}$$

where λ is the removal rate, m_f is the mass fraction remaining in the containment, H is the spray drop height, and Q is the spray water flux, calculated by dividing the spray flow rate (**F**) by the wetted cross-sectional area of the sprayed portion of the containment. The wetted cross-sectional area is determined by multiplying the containment cross-sectional area (**A**) by the sprayed fraction (**S_F**). The inner radius of containment is 52.5 ft, yielding a cross-sectional area of 8.659E3 ft^2. The first equation above is used to calculate the removal rate corresponding to a mass fraction of 0.9. Substituting this value into the second equation yields the removal for a given value of mass fraction. Since the removal rate is dependent on drop height and spray rate, the smallest (most conservative) value for each is used to calculate the lowest removal rate that will be applied over the entire time when sprays are credited.

Spray flux is derived as follows:

$$\mathbf{Q} = (\mathbf{F}\ \text{gpm})\ \underset{\text{[conversion]}}{(6.791\text{E-}2)}\ /\ (\mathbf{A}\ \text{ft}^2 \times \mathbf{S_F})$$

Table 3.2-4 presents the spray fall height of 116.5 ft which was derived by subtracting the 649'-6" elevation of the refueling floor from the weighted-average spray header elevation of 766'-0" (Reference 24), assuming one train of spray pumps available.

NUREG/CR-5966 [Page 170] recommends that for a volume with continuing source, the spray removal constant associated with a mass fraction of 0.9 be used until the time-dependent source terminates. Hence, the mass fraction should be assumed to remain at 0.9 from the start of the sprays until the end of the early in-vessel release phase at 1.8 hr. The resulting spray removal coefficient for $m_f = 0.9$ is 2.855 hr^{-1}, rounded down conservatively to **2.8 hr^{-1}**.

The spray removal coefficient of 2.8 hr^{-1} was used over the period that sprays are credited. The airborne inventory does not drop to 2 percent of the total particulate iodine released to the containment (i.e., a DF of 50) before spray termination at 0.91 hours.

Table 3.2-5 lists the aerosol and elemental iodine removal coefficients determined for KPS.

3.2.4.2 Natural Deposition

A reduction in airborne radioactivity in the containment by natural deposition within containment is credited. The model used is described in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," (Reference 25) and is incorporated into the RADTRAD-NAI computer code. This model is called the Powers model, set for the 10th percentile.

3.2.5 LOCA Analysis Assumptions & Key Parameter Values

3.2.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases to the EAB, LPZ, and Control Room resulting from a LOCA at Kewaunee Power Station (KPS).

RADTRAD can model a variety of processes that can attenuate and/or transport radionuclides. It can model sprays, filtered flow, and natural deposition that reduce the quantity of radionuclides suspended in the containment or other compartments. The RADTRAD models used in this calculation include the following pathways:

- Activity from the failed fuel enters the containment and is released to the atmosphere through containment leakage. All nuclides are released through this pathway. This pathway is not filtered.
- Containment air enters the shield building. A portion of the shield building air volume is discharged to the environment as necessary to maintain a negative pressure. Releases from the shield building to the environment are filtered.
- Negative pressure in the shield building is established within 10 minutes of the accident. During the first 10 minute interval, no credit is taken for filtering the shield building exhaust.
- Containment air enters the auxiliary building Special Ventilation (SV) zone to the environment. Releases from the SV zone to the environment are filtered.
- Activity in the sump leaks out of containment via the ECCS system and is released to the auxiliary building SV zone and then to the environment. Only iodine is released through this pathway. This pathway is filtered.
- Activity in the ECCS back-leaks to the RWST. The RWST vents into the auxiliary building and is captured by SV before exhausting to the environment. This pathway is filtered.

The revised LOCA analysis contains some changes to the plant specific assumptions and methods. These changes include:

- Conservative increase in core radionuclide curie inventory by applying a 1.06 multiplier to account for fuel management variations
- Conservative recalculation of spray removal coefficients based on a reduced spray droplet fall height
- Replacement of a sedimentation removal coefficient of 0.1 hr^{-1} with the Powers model built into RADTRAD
- Recalculation of offsite and control room X/Q dispersion factors
- Conservative increase in assumed iodine evolution rate from ECCS leakage to 10% for the entire 30-day duration of leakage. Current analysis of record assumes 10% evolution for 3 hours post accident, then 1% thereafter.
- Replacement of the assumed 1% iodine evolution rate from RWST back-leakage to a conservative DF=100.

The combined effect of these changes result in changes to the EAB, LPZ, and control room doses due to a KPS design basis LOCA. In all cases, the doses fall within required limits.

3.2.5.2 Basic Data & Assumptions for LOCA

Changes have been made to the AST LOCA. Tables 3.2-4 and 3.2-5 provide a complete list of inputs and assumptions used to reanalyze the KPS LOCA.

Table 3.2-4 Spray Removal Calculation Parameters			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Elemental Iodine Removal Coefficient			
K_g Gas phase mass transfer coefficient	3 m/min	No change	
T Spray drop fall time	13 seconds	9 seconds	Shorter fall height
F Volume flow rate of sprays	1148 gpm = 9,208 ft^3/hr	No change	
V Containment sprayed volume	1.32E6 ft^3	No change	
D Mass-mean diameter of the spray drops	1210 μm = 3.97E-3 ft	No change	
Particulate Iodine Removal Coefficient			
H Fall Height of droplets	150 ft	116.5 ft	Average spray header height to the refueling floor.
Q Spray Water Flux	Not Applicable	Derived value = $F/(A * S_F)$ 9.003E-3 cm^3 H_2O/cm^2-s	New method employed: NUREG/CR-5966
A Cross sectional area of containment	8.659E3 ft^2	No change	
S_F Sprayed fraction in containment	1.0	No change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
Core Power (MWt)	1782.6 (Licensed power of 1772 MWt with 0.6% uncertainty)	No Change	
Core Inventory (curies)	Licensed Uprated Core based on 1782.6 MWt multiplied by 1.03 to account for fuel management variations	Licensed Uprated Core based on 1782.6 MWt multiplied by 1.06 to account for fuel management variations	Conservative assumption
Dose Conversion Factors CEDE	Values are from Table 2.1 of Federal Guidance Report (FGR) 11	No Change	
Whole Body	ICRP 30 [Westinghouse TITAN5 code]	Table III.1 of FGR 12	Per RG 1.183
Core Release Fraction, Gap Release Fractions and Release Timing	Values from Table 2 and 4 of RG 1.183	No Change	
Initial Iodine Species in Containment (%) Elemental Methyl (organic) Particulate (aerosol)	 4.85 0.15 95	No Change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
	Containment		
Containment Leak Rate (wt%/day) 0-24 hours >24 hours	 0.2 0.1	No Change	
Containment Leak Path Fractions <u>0-10 minutes</u> Through Shield Bldg Through Aux Bldg SV Direct to Environment <u>10 minutes – 30 days</u> Through Shield Bldg Through Aux Bldg SV Direct to Environment	 0.0 0.10 0.90 0.89 0.10 0.01	No Change	
Shield Building Drawdown Time: (Tech Specs)	10 minutes	No Change	
Containment Volume (ft^3)	1.32E6	No Change	
Containment Purge Release Prior to Containment Isolation	Not Analyzed	Negligible	KPS is a licensed leak-before-break LBB plant (Reference 9). Per RG 1.183, the onset of gap release can be credited with a 10 minute delay for LBB. Containment purge isolation

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
			occurs within 37 seconds. Therefore, dose contribution from only TS RCS inventory is insignificant.
	Containment Sump and Sprays		
Iodine Chemical Form in the Sump (%)	100% Elemental	97% Elemental 3% Organic	Per RG 1.183
Containment Sump pH:	at least 7	No Change	
Containment Sump Volume (gal)	315,000	311,000	Subtracted tank volume measurement uncertainties
Containment Spray Coverage (%)	100	No Change	
Containment Spray Duration (hr)	0.917	0.91	Based on revised RWST low level signal based on minimum drain down time
Containment Spray Recirculation	No credited	No Change	
Containment spray Removal Coefficient (hr^{-1}) Elemental Particulate	 20 4.5	 15 2.8	Reduction in coefficient values is primarily a result of reduced droplet fall height

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Natural deposition (hr^{-1})	0.1	Power's Model set at the 10^{th} percentile	Per RG 1.183
Shield Building			
Shield Building Annulus Volume (ft^3)	3.74E+05	No Change	
Shield Building Participation Fraction	0.5	No Change	
Shield Building Ventilation and Recirculation Iodine Filter Efficiency (%) Elemental Methyl (organic) Particulate (aerosol)	 95 (includes safety factor of 2) 95 (includes safety factor of 2) 99	No Change	
Shield Building Air Flow to Environment (cfm) 0-10 min 10-30 min >30 min	 0 6600 3100	No Change	
Shield Building Recirculation Flow (cfm) 0-30 min >30 min	 0 2300	No Change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Auxiliary Building			
Participation with Auxiliary Building Volume or Hold-up	None	No Change	
Auxiliary Building Special Ventilation Iodine Filter Efficiency (%) Elemental Methyl (organic) Particulate (aerosol)	 95 (includes safety factor of 2) 95 (includes safety factor of 2) 99	No Change	
ECCS			
ECCS Leak Rate to Auxiliary Building (gal/hr)	12 (twice the leakage limit)	No Change	
ECCS Iodine Airborne Evolution (%) 0-3 hour >3 hour	 10 1	 10 10	Conservative change using RG 1.183 guidance
Plate-out in Aux Bldg (%)	50	No Change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Start of Recirculation (hr)	Conservatively assumes leakage starts at 0 hours	No Change	
RWST			
RHR Back-Leakage to RWST (gpm) 0-24 hour 1-30 day	 3 1.5	No Change	
Start of Back-Leakage (hr)	Conservatively assumes leakage starts at 0 hours	No Change	
RWST Iodine Airborne Evolution	1%	DF=100	Change in methodology consistent with and approved at other Dominion facilities (e.g., Millstone Unit 3 and North Anna). DF of 100 is conservative compared to a calculated DF greater than 300 for the KPS RWST.
Plate-out in Aux Bldg (%)	50	No Change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Core Total Iodine Mass (kg)	Not Used	14.23	Partition Coefficient in RWST is dependent upon total iodine. Value is conservatively based on ORIGEN results; Mass ratio of Total Iodine to Iodine-131 equals 35.
Maximum Sump Total Iodine Concentration (mg/liter)	Not Used	4.83	Based on sump volume and 40% of the core total iodine. RWST total iodine concentration is conservatively maximized to minimize iodine partition coefficient.
Maximum RWST Total Iodine Concentration (mg/liter)	Not Used	3.05	Maximum RWST total iodine concentration is achieved at 720 hours. Maximum concentration results in lowest partition coefficient (PC), from Ref. 26: 3.05 mg/l results in a PC=581.
RWST Tank Volume (gal)	Not Used	272,500 [3.64E4 ft^3]	RWST DF is a function of tank liquid and air volumes
Applied RWST DF	Not Used	100	Conservative value

Table 3.2-5 Basic Data and Assumptions for LOCA

Parameter or Assumption	CLB Value	Proposed Value		Reason for Change
Time Dependent RWST Liquid Volume from Back-Leakage	Not Used	Time (hrs) 0 3 6 12 24 48 96 200 400 720	Liquid (ft^3) 5253 5325 5397 5542 5830 6119 6697 7948 10354 14204	Calculated Values. New partition coefficient method requires calculation of RWST volumes and concentrations.
Calculated Time Dependent RWST DF	Not Used	Time (hrs) 0 3 6 12 24 48 96 200 400 720	DF 841 846 818 758 626 525 413 336 338 372	Calculated DF values consider time dependent RWST liquid and air volume and increasing iodine concentrations in the RWST. Over the entire 30 day accident, calculated DF values are greater than a factor of three above the applied DF value used in the RWST release analysis.
EAB X/Q (sec/m^3) 0 – 2 hr	 2.232E-04	 1.76E-04		New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
Control Room			
Control Room Volume (ft^3)	127,600	No Change	
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,500 (nominal)	2,750	Maximum flow considering +/- 10% uncertainty
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
CRPARS Filter Efficiency (%) Elemental Organic Particulate	90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	No Change	

Table 3.2-5 Basic Data and Assumptions for LOCA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Control Room Isolation (sec)	120	0	Control room isolation damper takes 10 seconds to close upon receipt of SI signal which is generated within seconds of the LOCA. Due to AST release delay of 30 seconds per RG 1.183, the control room will be isolated prior to any radioactive release.
CRPARS Start (sec)	120	133	Based on 10 second delay to switchover from normal ventilation to emergency operation, 63 second delay in diesel loading of CRPARS, and 60 seconds to open recirculation dampers
Control Room Unfiltered Inleakage (cfm)	800	No Change	Maximum (ASTM) E741 tracer gas test in Dec 2004 was 447±51 cfm (Ref. 20)

Table 3.2-5 Basic Data and Assumptions for LOCA

Parameter or Assumption	CLB Value	Proposed Value	Reason for Change
Release point(s)	Containment / Shield Bldg Rx Building Stack Exhaust Aux Building Stack Exhaust	No Change	
Control Room X/Q (sec/m³) Containment / Shield Bldg Rx Bldg Stack Exhaust Aux Bldg Stack Exhaust	for all releases 0 - 8 hrs 2.93E-03 8 – 24 hrs 1.73E-03 24 – 96 hrs 6.74E-04 96 – 720 hrs 1.93E-04	<u>0 – 2 hr</u> 1.74E-03 3.97E-03 2.90E-03	New ARCON96 control room X/Q estimates (Table 1.3-4) Prior to plume arrival, normal control room intake will isolate. X/Q values represent the worst case unfiltered inleakage location. For period values out to 720 hours, see Table 1.3-4

3.2.5.3 LOCA Containment Leakage Model

Containment leakage consists of filtered and bypass leakage. The total containment leak rate (L_a) is 0.2% per day (weight %/day) for the first 24 hours. Thereafter, the leak rate is one half or 0.1% per day for the remaining accident duration, out to 30 days.

For the first 10 minutes following the LOCA, the Shield Building is ignored while it is pumping down to vacuum conditions. Releases from containment are split, 10% being released from the filtered Auxiliary Building Ventilation (ABV) exhaust stack and the remaining 90% being released at ground level directly to the environment. The 10 minute interval is conservative because a measureable vacuum is developed in the shield building within 4 minutes of the fan startup.

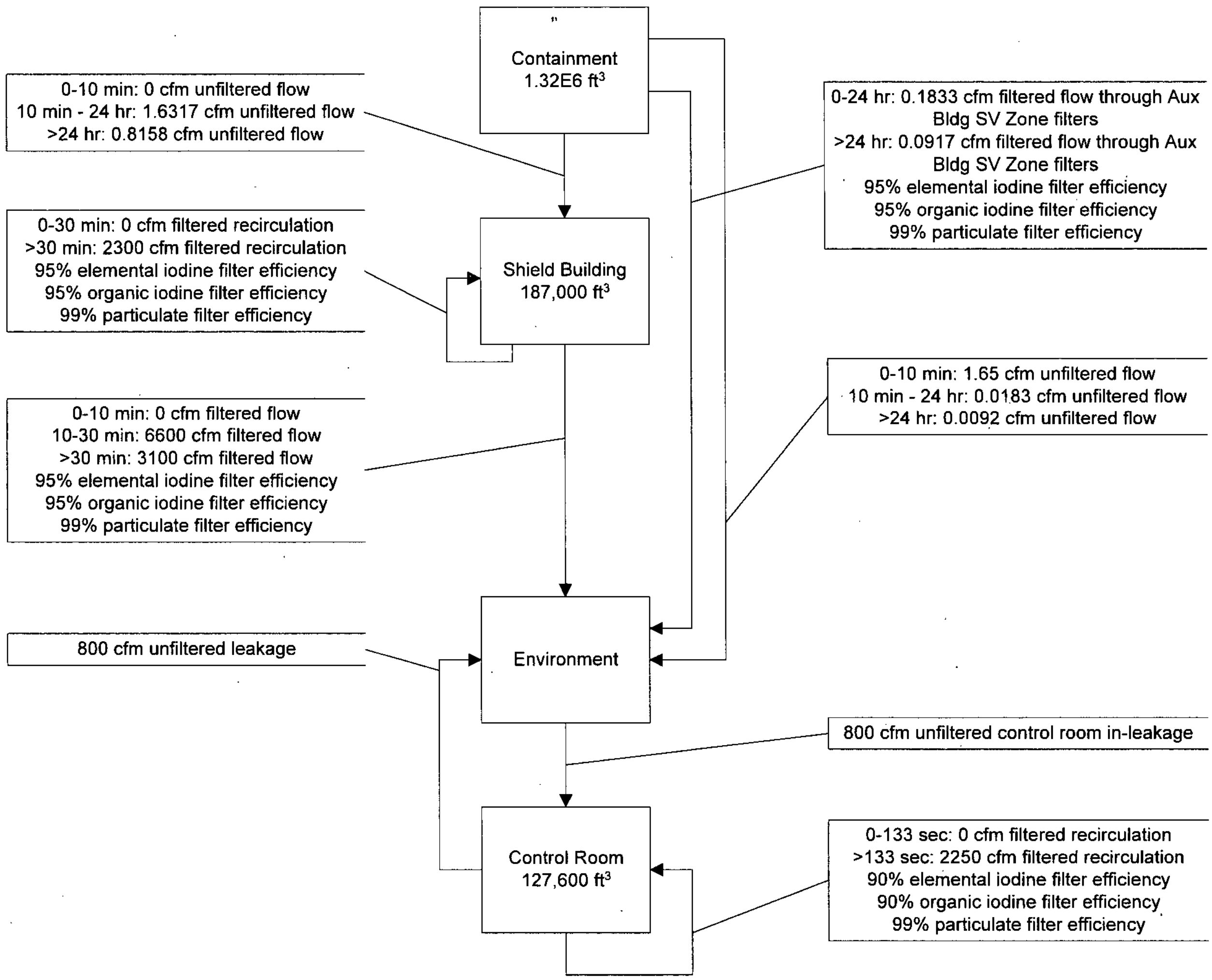
For the first 30 minutes, the recirculation of Shield Building annulus air is ignored. After vacuum conditions are achieved at 10 minutes, releases are assumed to begin out the filtered Shield Building Ventilation (SBV) exhaust stack at a conservatively high rate of 6600 cfm (highest starting drawdown rate prior to vacuum conditions). The percentage of bypass leakage assumed to escape directly to the environment is reduced to 1% of L_a with 10% continuing out the filtered ABV and the remaining 89% through the filtered SBV.

After 30 minutes post LOCA, Shield Building recirculation is credited. The split of containment releases remain 10%, 89% and 1% between the ABV, SBV and direct to the environment. The Shield Building requires an exhaust rate less than 2000 cfm to maintain vacuum conditions once achieved at 30 minutes, the analysis conservatively assumes 3100 cfm.

The collection, processing, and release of containment leakage vary depending on the location of the leak. Ventilation characteristics and release paths are different for the Shield Building and Auxiliary Building. KPS Technical Specification 5.5.14 leakage acceptance criteria provide the basis for release assumptions for containment leakage.

Figure 3.2-1 displays the assumptions, inputs and pathways used in RADTRAD to model KPS containment airborne releases from a design basis LOCA.

Figure 3.2-1 RADTRAD Model for Containment Airborne Releases



3.2.5.4 Model of ECCS Leakage

The Emergency Core Cooling System (ECCS) fluid consists of the contaminated water in the sump of the containment. This water contains 40% of the core inventory of iodine, 5% released to the sump water from the gap release phase and 35% released to the sump water from the early in-vessel phase. During a LOCA, the highly radioactive fluid is pumped from the containment sump to the recirculation spray headers and sprayed back into the containment sump. Also, following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS pumps from RWST to the containment sump.

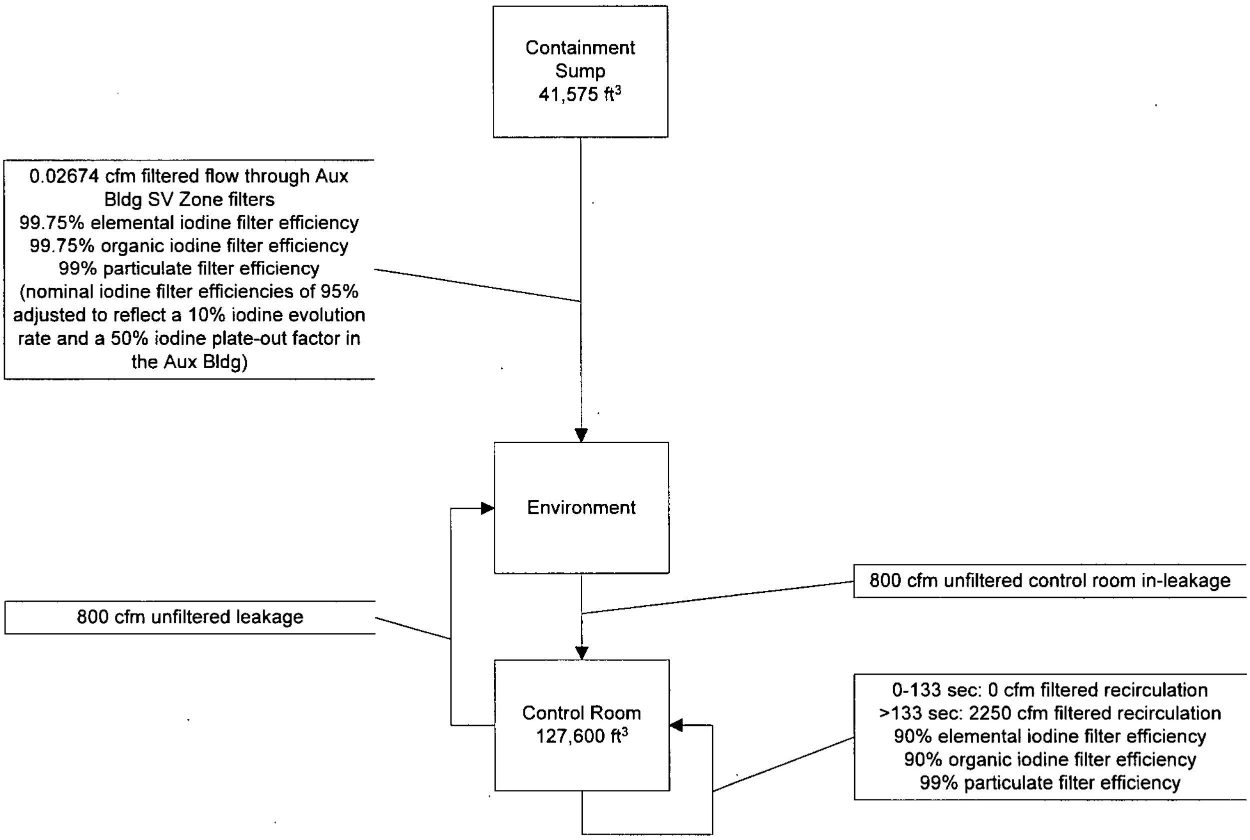
ECCS leakage develops when ESF (engineered safeguards feature) systems circulate sump water outside containment and leaks develop through packing glands, pump shaft seals and flanged connections. Station procedures specify a limit of 6 gallons per hour for total allowed ECCS leakage. In accordance with RG 1.183, the ECCS analysis makes use of two times the sum of the simultaneous leakage from all components in the ESF recirculation systems, or 12 gallons per hour for ECCS leakage. The leakage of recirculating sump fluids commences at 0.91 hours, which is the earliest time of recirculation. The analysis conservatively assumes leakage starts at 0 hours.

The temperature of the containment sump is conservatively assumed to reach a maximum of 293 degrees F (saturation conditions). At this maximum temperature, a flash fraction of less than 0.1 is calculated. Current analysis assumptions reduce the flash fraction to 0.01 after 3 hours when sump temperature drops below 212°F. However, per the guidance of RG 1.183, a conservative flash fraction of 0.1 is used for the ECCS leakage during the entire event for all sump temperatures. The water volume of the sump is 311,000 gallons and is assumed to remain constant.

Per KPS Licensing Basis, a 50% plate-out of iodine evolved from flashing ECCS fluid is credited on surfaces in the large Auxiliary Building volume. The xenon progeny from the iodine that plates out is included in the dose analyses. Dilution and decay within the Auxiliary Building volume are not credited.

Figure 3.2-2 displays the assumptions, inputs and pathways used in RADTRAD to model KPS ECCS Leakage from a design basis LOCA.

Figure 3.2-2 RADTRAD Model for ECCS Leakage into the Auxiliary Building



3.2.5.5 Model of ECCS Back Leakage to Refueling Water Storage Tank

Following a design basis LOCA, valve realignment occurs to switch the suction water source for the ECCS pumps from the Refueling Water Storage Tank (RWST) to the containment sump. This switch occurs when the level in the RWST reaches a defined setpoint and is modeled in RADTRAD-NAI as occurring at 0.91 hours following the initiation of the LOCA. In this configuration, MOV's and check valves in the normal suction line from the RWST and MOV's in the recirculation line provide isolation between this contaminated flow stream and the RWST. This RADTRAD-NAI analysis of the LOCA models leakage of ECCS fluid through these valves back into the RWST with subsequent leakage of the evolved iodine through the vent of the KPS RWST into the Auxiliary Building.

The RADTRAD-NAI source term used to model the ECCS leakage into the RWST contains only the iodine isotopes. This is because iodine is the only element in the containment sump water which was modeled as coming out of solution and becoming airborne. Forty percent of the core inventory of iodine isotopes were conservatively modeled as being instantaneously transported from the core to the containment sump. This iodine is modeled to be 97% in the elemental chemical form and 3% in the organic chemical form in accordance with RG-1.183.

The following two flowcharts shown in Figures 3.2-3 and 3.2-4 demonstrate the compartments and pathways used in RADTRAD to calculate the doses resulting from containment sump back-leakage into the RWST. Two separate models were used. The first models the RWST liquid space as a variable volume and was used to calculate doses due to the release of iodines. This model reduces the flow rate from the RWST to the environment to reflect the iodine partition coefficient in the RWST. Additionally, iodines released from the RWST vent into the Special Ventilation (SV) zone within the Auxiliary Building and get filtered prior to exhaust from the Auxiliary Building Stack. This model under-predicts the release of xenon isotopes produced from the decay of radioiodines in the RWST. The doses resulting from xenon are calculated using a second RWST release model.

Since the release of iodine is accounted for in the first model, a second model captures all iodine released from the sump in a 100% efficient iodine filter. All xenon resulting from iodine decay is released from the RWST out of the Auxiliary Building Stack. The combined doses resulting from the two RWST release models will conservatively predict the doses resulting from the iodine isotopes and their progeny.

Per KPS Licensing Basis, a 50% plate-out of iodine evolved from the RWST is credited on surfaces in the large Auxiliary Building volume. The xenon progeny from the iodine that plates out is included in the dose analyses. Dilution and decay within the Auxiliary Building volume is not credited.

The release scenario considers containment sump liquid leaking into the lines leading to the RWST and ignores any time delay that physically would occur as a result of contaminated sump fluid pushing "clean" fluid residing in the lines back to the RWST. Back-leakage to the RWST is conservatively assumed to start at 0 hours. The release of radioactivity is a result of partitioning between the contaminated fluid within the RWST and the air sitting above the fluid within the tank. To maintain an equal pressure within the tank, the amount of air released equals the volume of sump fluid that leaks into the tank. Over the 1 to 30 day period following the accident when 1.5 gallons per minute of sump fluid is assumed to leak into the RWST, the release of air is 0.201 ft^3/min.

A critical parameter in the radiological-impact analysis is the definition of a proper Partition Coefficient (PC) for the iodines in the RWST water. The PC applicable to the iodines in the RWST water was based on information in A. K. Postma, L. F. Coleman and R. K. Hilliard (Reference 26), "Iodine Removal from Containment Atmospheres by Boric Acid Spray," Report No. BNP-100, Battelle Memorial Institute, Pacific Northwest Laboratories (PNL), Richland, WA 99352 (7/1970). Use of BNP-100 is discussed in SRP 6.5.2 (Reference 22). For this application, the RWST is assumed to behave like a closed system for the establishment of equilibrium conditions between the water and air. This is the same method Dominion has employed at Millstone Unit 3 and North Anna submittals (References 33 and 34) for RWST releases due to sump back-leakage.

The critical factor in determining the magnitude of the PC is the total iodine concentration in the RWST water (on a mass basis, including stable iodine). Based on ORIGENS runs performed for Millstone Unit 3 and North Anna, which are both higher power PWRs (respectively 3650 MWt and 2940 MWt, compared to KPS at 1772 MWt), the mass of Total Iodine to Iodine-131 is 32.6 and 30.7, respectively. A conservative mass ratio of 35 was used to approximate the Total Iodine mass of 14.23 kg in the KPS core. Assuming 40% of the core iodine is released and conservatively contained in the sump, a sump iodine mass of 5.69 kg is assumed to maximize sump iodine concentration at 4.83 mg/liter. The maximum RWST iodine concentration of 3.05 mg/liter occurred at 30 days. The PNL report (Reference 26), shows how higher iodine concentration results in lower partitioning. The PC predicted at 3.05 mg/liter is approximately 581.

The iodine decontamination factors associated with the releases from the RWST were calculated using the relationship (from Reference 22, Standard Review Plan, Section 6.5.2):

$$DF = 1 + (V_{liq} / V_{air})\ PC$$

where V_{liq} and V_{air} are the liquid and air volumes between which the partitioning takes place. Using the smallest ratio of V_{liq} to V_{air} at the onset of back-leakage will predict the smallest predicted DF. Used in conjunction with the lowest PC of 581, applicable for the worst case tank concentration, a DF greater than 100 is obtained. A DF of 100 is typically employed in many applications in the nuclear power industry and has previously been demonstrated as being conservative for RWST release applications for Millstone Unit 3 and North Anna.

Actual time dependent values of DF calculated for the RWST over the 30-day accident are shown in Table 3.2-6. These DFs exist at various times using calculated RWST iodine concentrations and ratios of V_{liq} to V_{air} to show that an assumed DF of 100 provides at least a factor of three conservatism over the entire accident. The modeling of RWST releases over the entire 30-day accident period credits only a PC value of 155, applicable to final RWST iodine concentrations and volumes that would yield a DF equal to 100. Modeling the entire release period with a constant partition coefficient provides additional conservatism beyond the factor of three already discussed. As Table 3.2-6 shows, actual partitioning in the tank can be more than an order of magnitude greater.

Using the methodology employed at Millstone Unit 3 and North Anna to model RWST back-leakage releases, the application of the model incorporates numerous conservatisms to assure predicted RWST radioactive releases are adequately bounding for the KPS LOCA analysis.

Table 3.2-6 RWST Time Dependent DF Values

Parameter	Value				
DF Determination (RWST Liquid volume, RWST Air volume, PC and DF)	Time (hrs)	Liquid (ft^3)	Air (ft^3)	PC*	DF**
	0	5253	31180	4982	841
	3	5325	31108	4936	846
	6	5397	31036	4699	818
	12	5542	30892	4218	758
	24	5830	30603	3279	626
	48	6119	30314	2597	525
	96	6697	29737	1831	413
	200	7948	28485	1199	336
	400	10354	26079	849	338
	720	14204	22229	581	372

* Partition Coefficients taken from PNL Report [Reference 26] based on RWST total iodine concentration

** SRP 6.52 [Reference 22], DF = 1 + (V_{liq} / V_{air}) PC

Figure 3.2-3 RADTRAD Model for Iodine Back-Leakage into the RWST

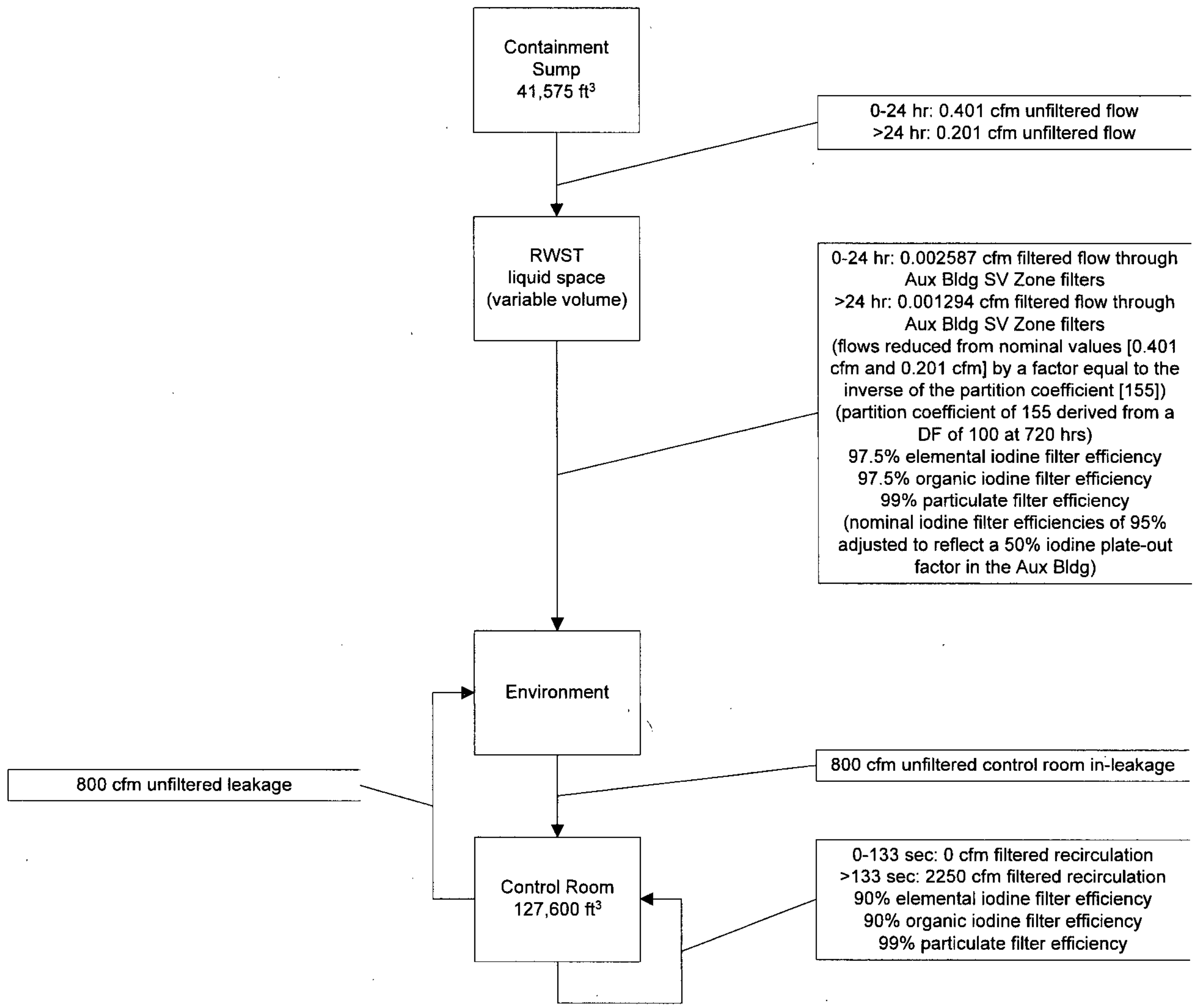
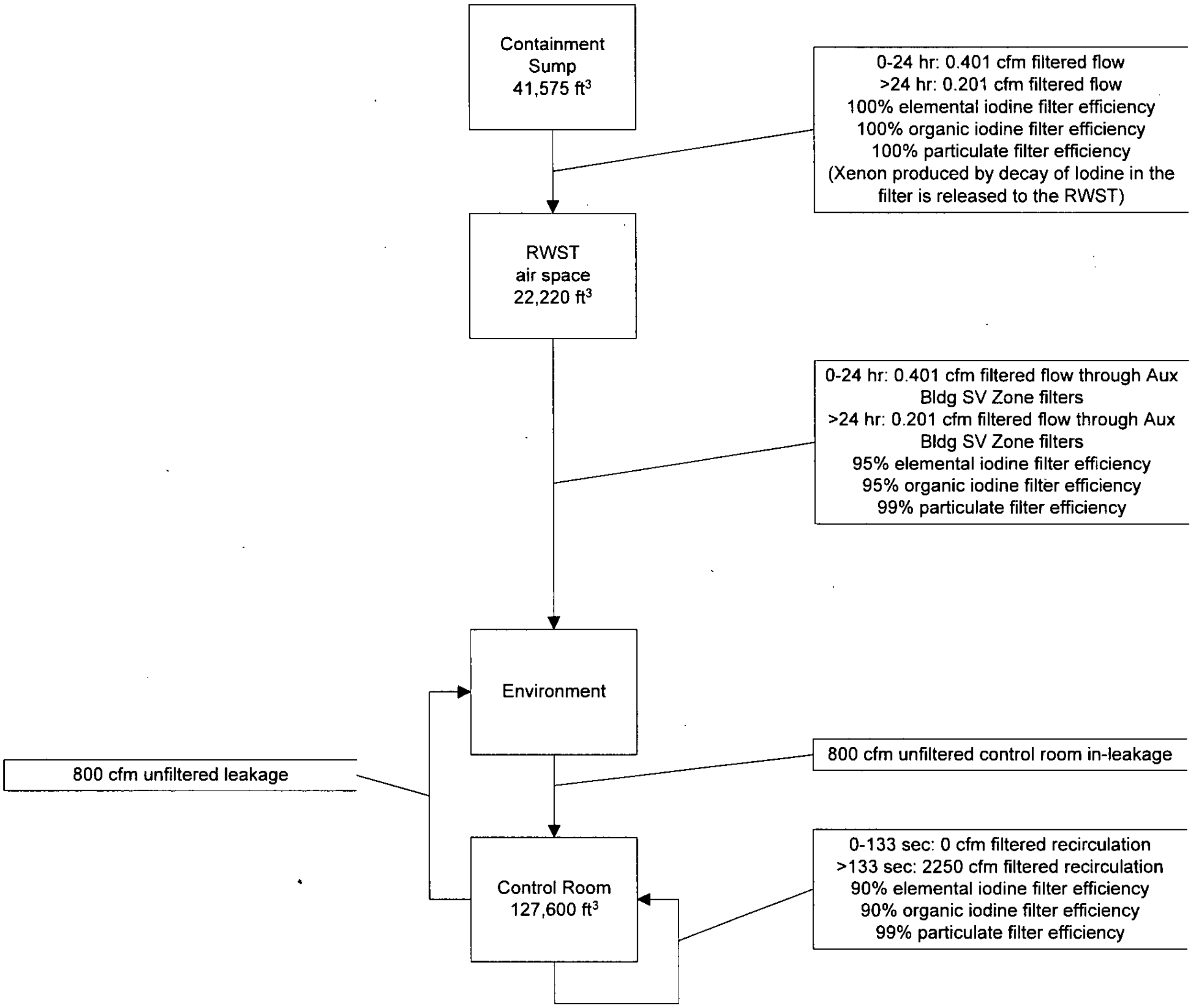


Figure 3.2-4 RADTRAD Model for Noble Gas Leakage from RWST



3.2.5.6 KPS Control Room

The control room volume is 127,600 ft^3. The LOCA causes a Safety Injection (SI) signal, which also isolates the control room (per current Licensing Basis). The control room is isolated within 10 seconds after the SI signal. Based on RG 1.183, the onset of the gap release does not start until 30 seconds post-LOCA. Therefore, the control room will be isolated prior to the arrival of the radioactive release.

Control room parameters are provided in Tables 1.3-1, 1.3-5, and 3.2-5. These parameters include the normal operation flow rates, the emergency operation flow rates, control room volume, filter efficiencies and control room operator breathing rates. In the analyses presented in this report, the control room is modeled as a discrete volume. The Table 1.3-4 atmospheric dispersion factors are calculated to determine the activity available for intake into the control room from releases. The inflow to the control room and the control room recirculation flow are used to calculate the activity introduced to the control room and cleanup of activity from that flow. The control room filter efficiencies are conservatively assumed at 90% for both elemental and organic and 99% for aerosol iodine.

The CR ventilation system provides a large percentage of recirculated air. Process radiation monitor channel R-23 monitors control room ventilation air for radiation. If a high radiation condition exists, the monitor initiates closure of the outside air intake and starts the CR post accident recirculation system (CRPARS). KPS control room isolation and start of CRPARS also occurs on either a Safety Injection or Steam Exclusion signal. In addition, local CR area radiation monitor channel R-1 monitors CR air for radiation and alarms when it reaches the CR area radiation monitor setpoint. No credit is taken for the alarms or automatic actions from R-23 and R-1 in the design basis LOCA.

The post LOCA dose consequences to the KPS control room are due to the following sources:

- Containment leakage
- ESF leakage
- RWST backflow

External shine sources are negligible to the overall control room dose consequences due to control room structure boundaries, penetration pathways and internal shield walls consisting of at least, or equivalent to, 18 inches of concrete (Reference 21).

3.2.6 LOCA Results

Table 3.2-7 lists TEDE to the EAB and LPZ from a LOCA at KPS. The dose to the EAB and LPZ is less than the 25 rem TEDE limit stated in 10 CFR 50.67 and Regulatory Guide 1.183. The EAB dose represents the worst 2-hour dose for each release pathway.

The dose to the KPS control room is less than the 5 rem TEDE limit specified in 10 CFR 50.67 and Regulatory Guide 1.183.

Table 3.2-7 Dose summary for a Kewaunee LOCA

Location	TEDE (rem)	Limits (rem)
EAB	0.5	25
LPZ	0.5	25
Control Room	4.1	5

3.3 Fuel Handling Accident (FHA)

This section describes the methods employed and results of the Fuel Handling Accident (FHA) design basis radiological analysis. The analysis includes doses associated with release of gap activity from a fuel assembly either inside containment or in the Spent Fuel Pool. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ) boundary, and in the KPS control room. The methodology used to evaluate the control room and offsite doses resulting from the FHA is consistent with RG 1.183 in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, PAVAND based EAB and LPZ atmospheric dispersion factors, and Federal Guidance Reports No. 11 and 12 dose conversion factors. Isolation of the control room prior to movement of recently irradiated fuel assemblies and manual operator action to initiate the CRPARS within 20 minutes of the release will be new requirements.

Amendment 190 for Kewaunee Power Station, dated March 8, 2007 (Reference 14), represents the current licensing basis for the FHA.

3.3.1 FHA Scenario Description

The design basis scenario for the radiological analysis of the FHA assumes that cladding damage has occurred to all of the fuel rods in one dropped fuel assembly. The rods are assumed to instantaneously release their fission gas contents to the water surrounding the fuel assembly. The analyses include the evaluation of FHA cases that occur in both the containment and the spent fuel pool (SFP). All radioactivity released from the damaged fuel is released over a two hour period. Release pathways considered include:

1. Spent fuel pool via the Aux. Bldg stack
2. Spent fuel pool via the roll-up doors
3. Containment personnel hatch to the Aux Bldg stack
4. Containment to the Reactor Building stack
5. Containment equipment hatch

A single KPS FHA scenario models the bounding FHA which does not credit mitigating systems (e.g., radiation monitor isolation, bypass and closure signals, or ventilation filtration) and maximizes source term, dispersion and dose. This bounding scenario provides the basis to allow all penetrations to be open under administrative control while moving recently irradiated fuel.

The results of this analysis show that control room isolation is required prior to moving recently irradiated fuel assemblies in order to maintain operator dose within 5 rem TEDE. KPS is proposing to remove credit for the Control Room Ventilation radiation monitor R-23 providing control room isolation. The R-23 system is not safety grade and consists of a single radiation monitor. In addition, the isolation signal generated by R-23 will not assure the closure of all control room ventilation dampers needed to provide complete control room isolation. The current Fuel Handling Accident (FHA) uses and credits the R-23 radiation monitor for control room isolation. The basis behind crediting R-23 relies on arguments that Operations will take appropriate actions to isolate the control room if R-23 fails to perform its isolation function. Removing credit for R-23 requires an alternative means to ensure control room isolation. The FHA requires that the control room be isolated prior to moving recently irradiated fuel and that manual operator action be taken to initiate the CRPARS within 20 minutes of the release.

3.3.2 FHA Source Term Definition

In accordance with Regulatory Position 3 of RG 1.183, the core source term was previously calculated using the ORIGEN2 code for a Stretch Power Uprate (SPU) to 1772 megawatt thermal (MWt) and used in Amendment No. 172, issued February 27, 2004 (Reference 11). The core curies include a 6% increase to account for fuel management variations (493.6 ± 10% EFPD, average enrichment of 4.5 w/o ± 10%, and core mass of 49.1 MTU ± 10%). This core inventory is described in the LOCA scenario (Section 3.2.2) and is used for the FHA with 100-hours of decay.

For the FHA analyses, the core inventory was used to calculate the gap activity of one fuel assembly for input to RADTRAD-NAI. The amount of fuel damage is the same whether the FHA is in the spent fuel pool or containment. Therefore, the only variable between a FHA in the containment or spent fuel pool is the release point. As with previous AST submittals, the FHA analysis assumes the resulting chemical form of the radioiodine in the water is 99.85% elemental iodine and 0.15% organic iodide.

3.3.3 FHA Release Transport

The FHA scenario does not credit operability or operation of the Spent Fuel Pool Sweep System nor does it credit any ventilation filtration systems or automatic functions. It is assumed that containment penetrations, (e.g., personnel hatch, equipment hatch, or other penetration) remain open for the duration of the 2-hour release. Modeling the release with the highest estimated control room X/Q from all possible release points and all possible intake points (normal intake and inleakage locations) maximizes the control room dose and represents the worst source to receptor orientation.

Releases from a FHA to the environment are at a rate of 3.454 air changes per hour. This assures that greater than 99.9% of the activity is released within 2 hours. In addition, the release rate is conservatively biased to release > 80% of all activity within the first half hour of the event. No credit is taken for dilution or mixing of the activity released to the Auxiliary Building or Containment air volumes.

All possible release pathways were considered from a FHA in either the SFP or containment. The most conservative pathway to the Control Room was modeled. The bounding pathway is an unfiltered release from the Reactor Building Ventilation Exhaust Stack which has the largest calculated control room X/Q (see section 3.1.1). The EAB and LPZ dispersion factors encompass all possible release points (see Section 3.1.2), and therefore are bounding.

3.3.4 FHA Atmospheric Dispersion Factors

3.3.4.1 FHA Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The FHA Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source points:

- Reactor Building Exhaust Stack
- Containment Equipment Hatch
- Auxiliary Building Exhaust Stack
- Fuel Area Roll-up Door

3.3.4.2 FHA Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.3.5 FHA Analysis Assumptions & Key Parameter Values

3.3.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a FHA at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

RADTRAD can model a variety of processes that can attenuate and/or transport radionuclides. The RADTRAD models used in the FHA calculations include the following:

- The damaged fuel assembly has been operating at the highest fuel rod power level. This conservative assumption maximizes fuel gap activity and dose consequences.

- All fuel rods in the dropped assembly fail, instantaneously releasing activity contained in the fuel gap into the water that the assembly is being moved within.
- The overall pool decontamination factor (DF) for iodine is 200 (see Section 3.3.5.3).
- 25% of the fuel rods in the worst peak assembly do not comply with footnote 11 in RG 1.183 (Reference 1). Higher gap fractions applicable to the FHA and previously approved in KPS License Amendment 190 on March 8, 2007 (Reference 14) are assumed in these rods (see Table 3.3-1).
- 75% of the fuel rods in the worst peak assembly meet the criteria in RG 1.183 footnote 11 and use the suggested gap fractions for non-LOCA events.
- Iodine leaving the water is 57% elemental and 43% organic per RG 1.183. All noble gases release instantaneously to the air above the water.
- All activity released from the water surface is released to the environment within a 2-hour period without credit for mixing or dilution within the building volume.
- The maximum X/Qs from any applicable release point to either the control room intake or control room inleakage pathway is used throughout the entire 2-hour release.

The FHA approved in Amendment 190 (Reference 14) differs from the FHA in this amendment request by the following:

1. Revised Control Room X/Qs (based on ARCON96)
2. Revised Off-site X/Qs (based on PAVAND)
3. Control Room Inleakage Assumption decreased from 1500 cfm to 800 cfm
4. Percent of fuel rods in the dropped assembly that exceed the criteria set forth in footnote 11 of RG 1.183 decreased from 50% to 25%
5. Automatic isolation of the control room by the non safety-grade, non-redundant control room ventilation monitor R-23 is no longer credited
6. Control room isolation is required prior to moving recently irradiated fuel assemblies

7. All spent fuel pool area and containment penetrations (including the equipment hatch) are allowed to be open under administrative control during fuel manipulations
8. CRPARS is credited for operation within 20 minutes of the FHA based on operator action

The combined effect of these changes result in changes to the EAB, LPZ, and control room doses due to a KPS design basis FHA. In all cases, the doses fall within required limits.

Figure 3.3-1 displays the assumptions, inputs and pathways used in RADTRAD to model the KPS FHA.

3.3.5.2 Basic Data and Assumptions

Changes have been made to the AST FHA. Table 3.3-1 provides a complete list of inputs and assumptions used to reanalyze the KPS FHA.

Figure 3.3-1 RADTRAD Model for FHA

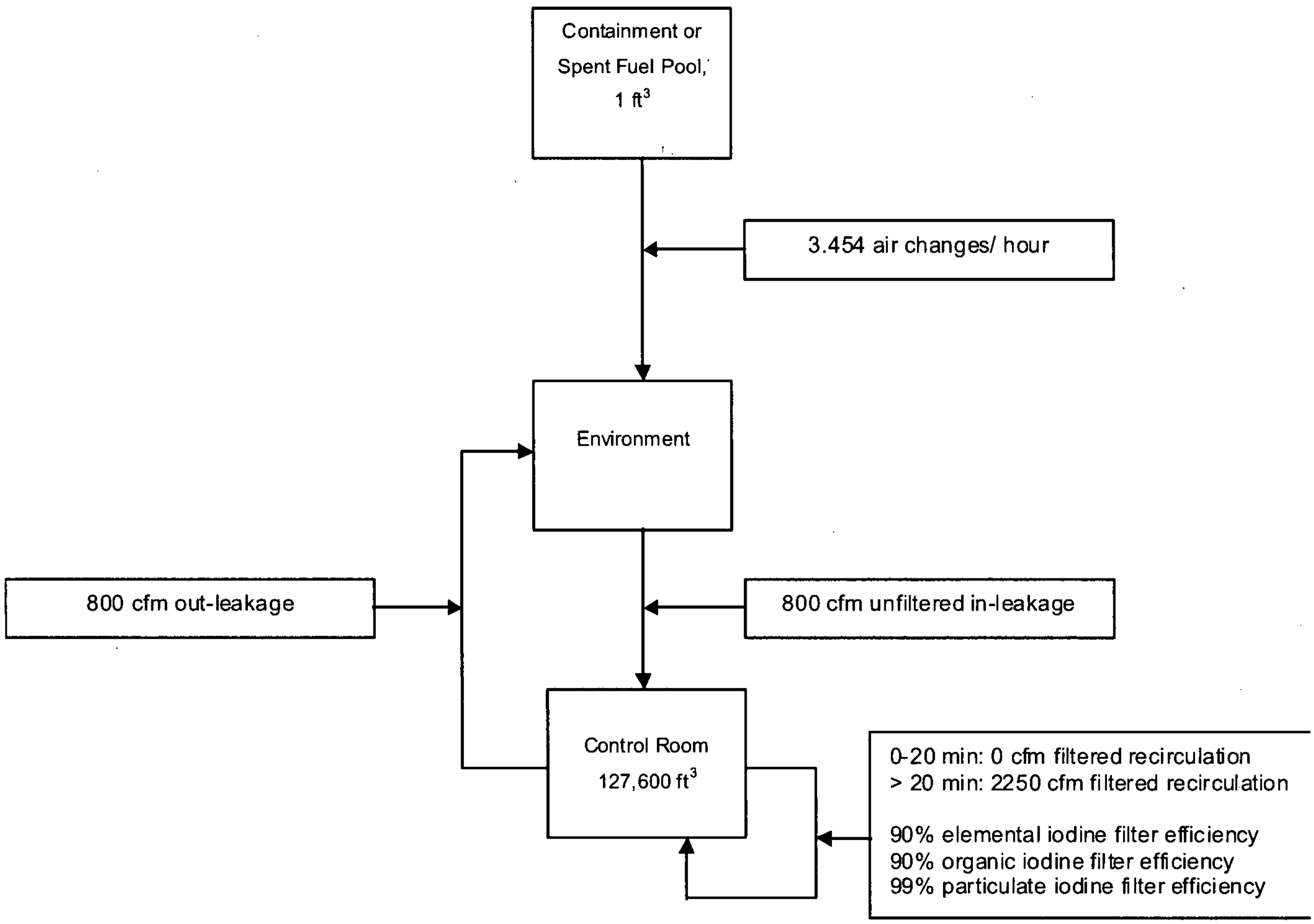


Table 3.3-1 Basic Data and Assumptions for FHA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
Fuel Damage:	1 assembly	No Change	
Decay Time:	100 hours	No Change	
Radial Peaking Factor:	1.7	No Change	
Duration of Release	2 hours	No Change	
Pool Decontamination Factor:	Noble Gases: 1 Iodines: 200 (effective DF)	No change	
Percentage of Fuel Rods that Exceed the Requirements of Footnote 11 of RG 1.183	50%	25%	Excess margin is being removed from analysis. This limit is reflected in the KPS blank Reload Safety Analysis Checklist and verified on a cycle specific basis. For rods above footnote 11 criteria, the gap fractions listed in Regulatory Guide 1.25 (as modified by the direction of NUREG/CR-5009) are used with the design peaking factor of 1.7.

Table 3.3-1 Basic Data and Assumptions for FHA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Gap Fractions Fuel that complies with footnote 11 of Regulatory Guide 1.183	I-131 0.08 Kr-85 0.10 Other 0.05 - noble gases - halogens	No Change	
Fuel that does not comply with footnote 11 of Regulatory Guide 1.1.83	I-131 0.12 Kr-85 0.30 Other 0.10 - noble gases - halogens	No Change	
Release Point:	Not applicable (One site control room X/Q represented any release point to the control room)	Reactor Building Exhaust Stack	Current control room X/Q is treated as the bounding X/Q from any release point to the control room. New ARCON96 analyses (Table 1.3-4) have been performed to analyze control room X/Qs from all release points.

Table 3.3-1 Basic Data and Assumptions for FHA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Activity in One Fuel Assembly	Nuclide / Activity (Ci): I-131 2.99E+05 I-132 2.53E+05 I-133 3.15E+04 I-135 2.25E+01 Kr-85m 2.22E-02 Kr-85 4.72E+03 Kr-87 4.75E-19 Kr-88 7.77E-06 Xe-131m 4.50E+03 Xe-133m 1.06E+04 Xe-133 5.75E+05 Xe-135m 3.60E+00 Xe-135 1.10E+03	No Change	
EAB X/Q (sec/m^3) 0 – 2 hr	2.232E-04	1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	Period / LPZ 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	Period / LPZ 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)

Table 3.3-1 Basic Data and Assumptions for FHA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Control Room			
Control Room Volume (ft^3)	127,600	No Change	
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
Open Penetrations	Personnel Hatch	ANY penetration will be allowed to be open under administrative control during movement of irradiated fuel	LAR request to allow ANY penetration to be open under Administrative Control
CRPARS Filter Efficiency (%) Elemental Organic Particulate	90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	No Change	
Control Room Unfiltered Inleakage (cfm)	1500	800	Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20)

Table 3.3-1 Basic Data and Assumptions for FHA

Parameter or Assumption	CLB Value	Proposed Value	Reason for Change
Control Room X/Q (sec/m^3)	2.93E-3	4.88E-03	New ARCON96 estimates of control room X/Q (Table 1.3-4) show the Rx Bldg Exhaust Stack has the highest dispersion factor to the control room of any applicable release pathway
Control Room Isolation (min)	1	0	CR ventilation intake rad monitor R-23 is no longer credited. Open penetration allowance will require the control room to be isolated prior to movement of recently irradiated fuel.
Control Room Post Accident Recirculation system (CRPARS) Start (min)	1	20	Operator action is required to start CRPARS within 20 minutes of event initiation based on communication with the refuel operator and radiation monitors going into alarm.

3.3.5.3 Decontamination Factor in Less than 23 feet of Water

Per Regulatory Guide 1.183, if the depth of water above a damaged fuel assembly is 23 feet or greater, the decontamination factors (DF) for elemental and organic species are 500 and 1, respectively, giving an overall effective DF of 200. Design configuration of a fuel assembly drop in the containment and spent fuel pool where examined to confirm the water depth of 23 feet. Based on the assumption that the fuel assembly will be horizontal once it comes to rest, it was determined that an assembly lying on the reactor vessel flange will have approximately 22.35 feet of water above the highest point of the assembly to the water surface. In the spent fuel pool, greater than 23 feet of water will exist.

The depth of 22.35 feet of water was evaluated to verify an effective decontamination factor of 200 using WCAP-7828 (Reference 27). Using the methods defined in the WCAP with conservative assumptions to minimize predicted decontamination factors for various depths of water, a DF of greater than 500 was determined for elemental iodine. The use of an overall effective DF of 200 was determined to be appropriate per RG 1.183.

3.3.5.4 Recently Irradiated Fuel Determination

The age of Recently Irradiated Fuel (RIF) was determined using an iterative approach to determine a decay time that results in a control room dose within the 5 rem limit without requirements for operability of control room emergency ventilation systems. Off-site dose analyses are unaffected by the determination of RIF. 375 hours was selected as the basis for the definition of RIF based on RADTRAD runs that were made to determine when control room dose is < 5 rem TEDE without crediting any control room emergency ventilation or operator action. The worst case dispersion factor of any applicable release pathway (i.e., Reactor Building Exhaust Stack X/Q = 4.88E-03 sec/m^3) was used in the control room dose model. The source term was determined by decaying the 100-hour decayed source term (net activity) from Table 3.3-1 by an additional 275 hours (for a total decay of 375 hours). The 375-hour decayed isotopic inventory used in the RADTRAD NIF file is listed below.

Nuclide	Net Activity (Ci)
I-131	8.61E+01
I-133	1.99E-03
Kr-85	1.22E+03
Xe-133m	3.37E+01
Xe-133	1.52E+04

Recently Irradiated Fuel definition will be based on 375 hours of decay, post-shutdown. The control room operator dose based on RIF results in less than 5 rem TEDE.

3.3.6 FHA Analysis Results

The offsite and control room doses are listed below. The KPS Fuel Handling Accident releases essentially all activity of one damaged fuel assembly over a two-hour release period.

The associated worst case TEDE for the FHA scenario is presented in Table 3.3-2. All doses are less than the limits specified in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 3.3-2 Dose Summary for the Fuel Handling Accident Analysis

Location	TEDE (rem)	Limits (rem)
EAB	0.6	6.3
LPZ	0.2	6.3
Control Room	4.3	5

3.4 Steam Generator Tube Rupture Accident

This section describes the methods employed and the results of the Steam Generator Tube Rupture (SGTR) design basis radiological analysis. This analysis included doses associated with the releases of the radioactive material initially present in primary liquid, secondary liquid and iodine spiking. Doses are calculated at the Exclusion Area Boundary (EAB) for the worst-case two-hour period, at the Low Population Zone Boundary (LPZ), and in the KPS Control Room. The methodology used to evaluate the doses resulting from a SGTR is consistent with RG 1.183 (Reference 1) and utilized Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

3.4.1 SGTR Scenario Description

A steam generator tube rupture (SGTR) is a break in a tube carrying primary coolant through the steam generator. This postulated break allows primary liquid to leak to the secondary side of one of the steam generators (denoted as the affected generator) with an assumed release to the environment through the steam generator Power Operated Relief Valves (PORVs). The PORV on the affected steam generator is assumed to open to control steam generator pressure at the beginning of the event, and remain open until operator action is taken to close the PORV within 55 minutes. Hence, the affected generator discharges steam to the environment for 55 minutes (0.92 hours) until the generator is isolated by closure of the steam generator PORV. Flashed and un-flashed break flow in the affected steam generator is assumed to continue for the duration of the 55 minute period.

The intact generator discharges steam for a period of 29 hours until the primary system has cooled sufficiently to allow a switchover to Residual Heat Removal System (RHRS) cooling. At 29 hours, the RHRS can remove all the decay heat to achieve cold shutdown and steaming is no longer required for cooldown. No fuel damage is predicted as a result of a SGTR. Therefore, consistent with the current licensing analysis basis, the SGTR analysis was performed assuming both a pre-accident iodine spike and a concurrent accident iodine spike. In addition, the impact of a coincident loss-of-offsite power (LOOP) at the time of tube rupture was considered. In accordance with Regulatory Guide 1.183, release of noble gases without credit for holdup has been analyzed.

3.4.2 SGTR Source Term Definition

Initial radionuclide concentrations in the primary and secondary systems for the SGTR accident are determined based on the maximum Technical Specification levels of activity. The SGTR accident analysis indicates that no fuel rod failures occur as a result of these transients. Thus, radioactive material releases were determined by the radionuclide concentrations initially present in primary liquid, secondary liquid, and iodine spiking. These values are the starting point for determining the curie input for the RADTRAD-NAI code runs.

Regulatory Guide 1.183 specifies that the released activities should be the maximum allowed by the Technical Specifications. Table 3.4-1 lists all the primary and secondary liquid radionuclide concentrations that are used in the analysis. Primary side concentrations are based on the proposed new Technical Specification 3.4.16 limits of 16.4 μCi/gm DE Xe-133 for gross gamma and 0.1 μCi/gm DE I-131 for iodine. Secondary side concentrations are based on the proposed new Technical Specification 3.7.16 limit of 0.05 μCi/gm DE I-131 for iodine. In addition, since there is not a Technical Specification limit for the secondary side gross gamma activity, activities in the steam generator liquid were derived by assuming one half of the primary side activity to ensure that a suitably conservative source term was used.

Table 3.4-1 Primary Coolant and Secondary Side Nuclide Concentrations

Nuclide	Primary (µCi/gm)	Secondary (µCi/gm)*
Kr-85m	4.76E-02	
Kr-85	2.37E-01	
Kr-87	3.12E-02	
Kr-88	9.04E-02	
Xe-131m	8.40E-02	
Xe-133m	9.49E-02	
Xe-133	6.67E+00	
Xe-135m	1.38E-02	
Xe-135	2.40E-01	
Xe-138	1.73E-02	
Br-83	2.51E-03	1.26E-03
Br-84	1.24E-03	6.20E-04
I-130	9.49E-04	4.75E-04
I-131	7.82E-02	3.91E-02
I-132	7.97E-02	3.99E-02
I-133	1.17E-01	5.85E-02
I-134	1.62E-02	8.10E-03
I-135	6.40E-02	3.20E-02
Rb-86	8.95E-04	4.48E-04
Rb-88	1.13E-01	5.65E-02
Rb-89	5.15E-03	2.58E-03
Cs-134	8.17E-02	4.09E-02
Cs-136	9.10E-02	4.55E-02
Cs-137	6.02E-02	3.01E-02
Ba-137m	5.70E-02	2.85E-02
Cs-138	2.65E-02	1.31E-02

Table 3.4-1 Primary Coolant and Secondary Side Nuclide Concentrations

Nuclide	Primary (µCi/gm)	Secondary (µCi/gm)*
Cr-51	5.40E-03	2.70E-03
Mn-54	4.00E-04	2.00E-04
Fe-55	2.10E-03	1.05E-03
Fe-59	5.10E-04	2.55E-04
Co-58	1.40E-02	7.00E-03
Co-60	1.30E-03	6.50E-04
Sr-89	1.15E-04	5.75E-05
Sr-90	5.73E-06	2.87E-06
Sr-91	1.54E-04	7.70E-05
Sr-92	3.43E-05	1.72E-05
Y-90	1.60E-06	8.00E-07
Y-91m	8.31E-05	4.16E-05
Y-91	1.55E-05	7.75E-06
Y-92	2.97E-05	1.49E-05
Y-93	9.85E-06	4.93E-06
Zr-95	1.80E-05	9.00E-06
Nb-95	1.80E-05	9.00E-06
Mo-99	2.10E-02	1.05E-02
Tc-99m	1.95E-02	9.75E-03
Ru-103	1.54E-05	7.70E-06
Ru-106	5.22E-06	2.61E-06
Rh-103m	1.53E-05	7.65E-06
Rh-106	5.22E-06	2.61E-06
Te-125m	1.91E-05	9.55E-06
Te-127m	8.64E-05	4.32E-05
Te-127	3.64E-04	1.82E-04

Table 3.4-1 Primary Coolant and Secondary Side Nuclide Concentrations

Nuclide	Primary (μCi/gm)	Secondary (μCi/gm)*
Te-129m	2.94E-04	1.47E-04
Te-129	3.82E-04	1.91E-04
Te-131m	6.91E-04	3.46E-04
Te-131	3.73E-04	1.87E-04
Te-132	8.10E-03	4.05E-03
Te-134	7.91E-04	3.96E-04
Ba-140	1.15E-04	5.75E-05
La-140	3.88E-05	1.94E-05
Ce-141	1.76E-05	8.80E-06
Ce-143	1.34E-05	6.70E-06
Ce-144	1.33E-05	6.65E-06
Pr-143	1.69E-05	8.45E-06
Pr-144	1.33E-05	6.65E-06

*** Secondary equals primary times 0.5.**

Regulatory Guide 1.183 stipulates that SGTR accidents consider iodine spiking above the value allowed for normal operations based both on a pre-accident iodine spike and a concurrent accident spike. For KPS, the maximum iodine concentration that will be allowed by the proposed Technical Specification 3.4.16 as the result of an iodine spike is 10 μCi/gm DE I-131. The spike limit is being lowered commensurate with the reduction in reactor coolant activity. The pre-accident iodine spike concentrations corresponding to 10 μCi/gm DE I-131 are listed in Table 3.4-2. Regulatory Guide 1.183 defines a concurrent iodine spike as an accident initiated value 335 times the appearance rate corresponding to the Technical Specification 3.4.16 limit for normal operation (0.1 μCi/gm DE I-131 RCS TS limit) for a period of 8 hours. The concurrent iodine spike appearance rates based on 335 times the 0.1 μCi/gm DE I-131 concentration are listed in Table 3.4-3. Appearance rates developed address the issues raised by NSAL-00-004 (Reference 28).

The dose conversion factors used to calculate the TEDE doses and DE I-131 for the Steam Generator Tube Rupture accident were taken from Table 3.2-3 for the isotopes required by Regulatory Guide 1.183 for the SGTR analysis.

Table 3.4-2 Pre-accident Iodine Spike RCS Concentration

Nuclide	Iodine Activity in RCS 0.1 DE I-131 μCi/gm	Iodine Activity in RCS 10 DE I-131 μCi/gm
I-131	7.82E-02	7.82E+00
I-132	7.97E-02	7.97E+00
I-133	1.17E-01	1.17E+01
I-134	1.62E-02	1.62E+00
I-135	6.40E-02	6.40E+00

Table 3.4-3 Concurrent Iodine Spike SGTR RCS Concentration

Nuclide	Appearance rate for 0.1 μCi/gm DE I-131 Ci/hr	Spike = 335 SGTR Appearance Rate Ci/hr
I-131	1.80E+00	6.02E+02
I-132	4.75E+00	1.59E+03
I-133	3.10E+00	1.04E+03
I-134	1.93E+00	6.45E+02
I-135	2.26E+00	7.57E+02

3.4.3 SGTR Release Transport

Affected Steam Generator

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the affected steam generator by the break flow. The break flow is terminated after 55 minutes when the generator is isolated by closure of the PORV. A fraction of the break flow is assumed to flash to steam in the affected generator and to pass directly into the steam space of the affected generator with no credit taken for scrubbing by the steam generator liquid. The radionuclides entering the steam space as the result of flashing pass directly to the environment through the Steam Generator PORVs. The remainder of the break flow enters the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the break flow are released as a result of secondary liquid boiling. A partition factor of 100 for all non-noble gas isotopes is assumed during boiling. Thus 1% of the iodines and particulates are released from the steam generator liquid to the environment along with the steam flow (moisture carryover is not actually modeled but is instead bounded by application of the partitioning factor). All noble gases are released from the primary system to the environment without reduction or mitigation. The transport model utilized for iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183.

Intact Steam Generator

The source term resulting from the radionuclides in the primary system coolant and from the iodine spiking in the primary system is transported to the intact generator by the leak-rate Limiting Condition for Operation (150 gallons per day) specified in Technical Specification LCO 3.4.13. All radionuclides in the primary coolant leaking into the intact generator are assumed to enter the steam generator liquid. Releases of radionuclides initially in the steam generator liquid and those entering the steam generator from the leakage flow are released as a result of secondary liquid boiling, including an allowance for a partition factor of 100 for all non-noble gas isotopes. Thus 1% of the iodines and particulates are assumed to pass into the steam space and then directly to the environment. All noble gases that are released from the primary system to the intact generator are released to the environment without reduction or mitigation. Releases were assumed to continue from the intact generator for a period of 29 hours after which the RHRS is credited for removing 100% of decay heat with no requirement for steaming to augment cooldown.

3.4.4 SGTR Atmospheric Dispersion Factors

3.4.4.1 SGTR Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The SGTR Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source points:

- "A" Steam Generator PORV
- "B" Steam Generator PORV

The control room X/Qs represent the highest values calculated based on the shortest distance measured from each applicable source location to control room receptor location (see Figure 3.1-1).

3.4.4.2 SGTR Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.4.5 SGTR Key Analysis Assumptions and Inputs

3.4.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a SGTR at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

There are several aspects of the SGTR analysis that require multiple RADTRAD models due to limitations of the code. This is due primarily to treatment of the source terms because noble gases are released without mitigation and iodines and particulates are released crediting partitioning and moisture carryover. The different models include:

- Pre-incident spike impact – iodine and daughters
- Pre-incident spike impact – noble gas
- Coincident spike impact – iodine and daughters
- Coincident spike impact – noble gas
- Initial RCS TS activity – iodine and particulate
- Initial RCS TS activity – noble gas
- Secondary side bulk liquid – iodine and particulate

A schematic shown in Figure 3.4-1 provides an overall picture of the SGTR releases to environment. Maximum and minimum values are provided for secondary side bulk liquid mass. The minimum value is used to reduce holdup for primary to secondary releases and the maximum value is used to maximize secondary side inventory; this is done to maximize dose from primary to secondary side releases.

3.4.5.2 Basic Data & Assumptions for SGTR

The Basic Data and Assumptions are listed below in Table 3.4-4. A time-line of events is provided in Table 3.4.5. Steam and break flow data are listed in Tables 3.4-6 to 3.4-8. Control room information is available in both Tables 3.4-4 and 1.3-1.

Figure 3.4-1 SGTR Radioactive Release Schematic

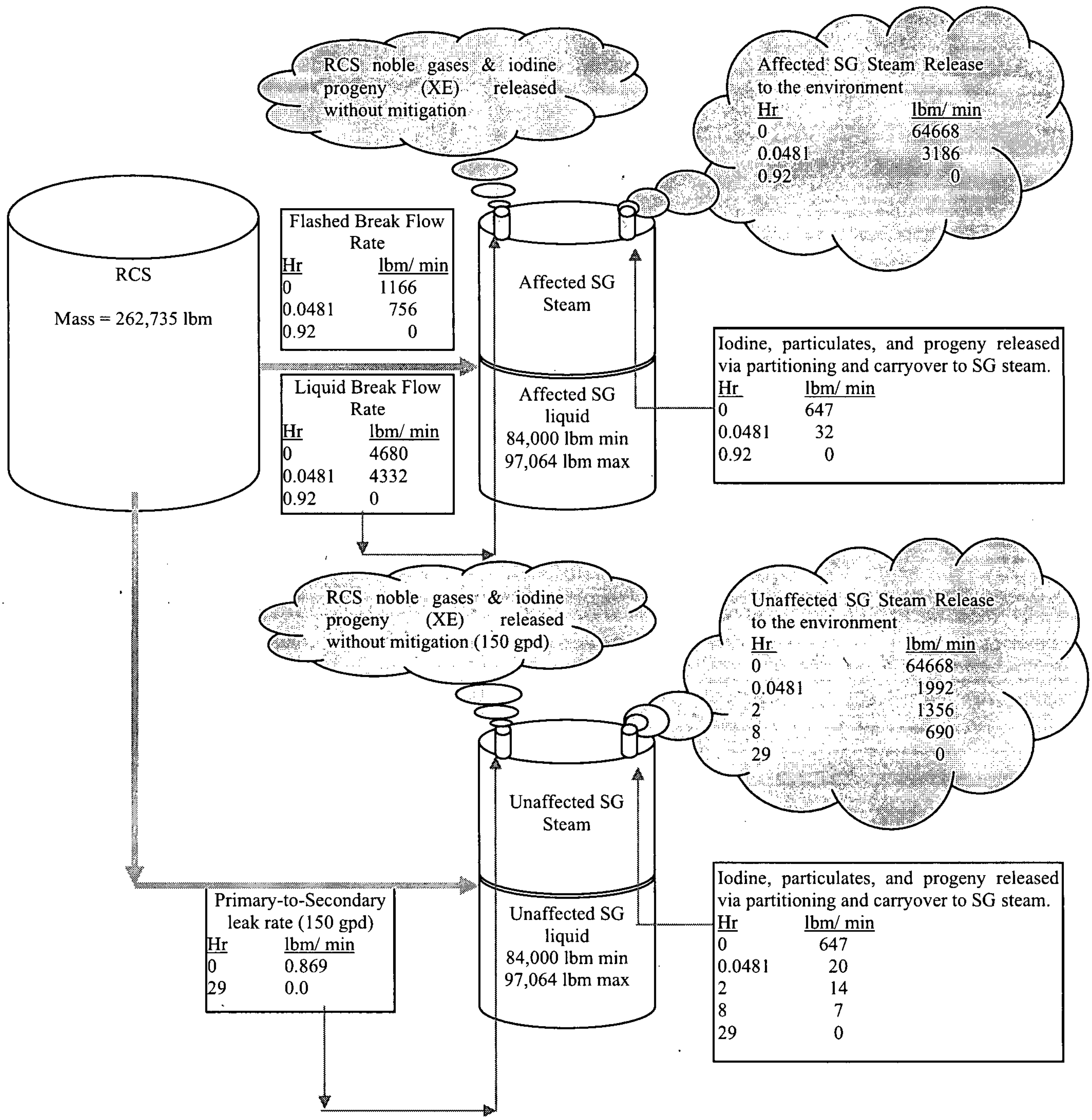


Table 3.4-4 Basic Data and Assumptions for SGTR			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
Primary Coolant Specific Activity Limit DE I-131 (μCi/gm) Gross Activity	 1 Not Included	 0.1 α 0.1 μCi/gm DE I-131	Technical Specification limits were reduced in order to maintain control room doses within acceptable limits. Derived from the 1% failed fuel inventory and equivalent to the failed fuel for the TS DE I-131 limit.
Primary Coolant Concentrations at TS Limit μCi/gm I-131 I-132 I-133 I-134 I-135	 7.80E-01 7.93E-01 1.16E+00 1.61E-01 6.37E-01	 7.82E-02 7.97E-02 1.17E-01 1.62E-02 6.40E-02	Current values include 5% variation to consider minor variations in fuel design (e.g., enrichment, core mass and cycle length). Proposed values are adjusted to allow 10% variation, to make consistent with similar allowance built into core inventory curies.
Primary Coolant Noble Gas Activity	595 μCi/gm DE Xe-133	16.4 μCi/gm DE Xe-133	The revised Noble gas limit corresponds to an equivalent level of fuel failure (0.027%) as the TS DE I-131 limit of 0.1 μCi/gm
Iodine Spike	500	335	Per RG 1.183
Accident-Initiated spike Duration (hr)	4	8	Per RG 1.183

Table 3.4-4 Basic Data and Assumptions for SGTR			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Iodine Appearance Rate I-131 I-132 I-133 I-134 I-135	Ci/min 0.301 0.788 0.519 0.319 0.377	Ci/hr 1.80 4.75 3.10 1.93 2.26	The difference in the iodine appearance rates reflects a unit conversion and a factor of ten reduction directly related to the reduced TS specific activity limit. Values on a consistent unit basis are shown to the right. Ci/min 0.030 0.079 0.052 0.032 0.038
Primary to Secondary Leak rate (gpd/SG)*	150	No Change	
Pre-Accident Spike Coolant Activity (μCi/gm DE I-131)	20	10	Proposed TS spike limit was lowered commensurate with primary coolant activity reduction.
Iodine Partitioning	PC for iodine = 100	No Change	
Iodine chemical form of Primary-to-Secondary Leakage (%)	Elemental 97 Organic 3 Particulate 0	No Change	
Moisture Carryover in Unaffected Steam Generators	1%	No Change	
Tube Uncovery.	No tube bundle uncovery assumed.	No Change	
Scrubbing of Flashed Break Flow	Not Credited	No Change	

Table 3.4-4 Basic Data and Assumptions for SGTR			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Secondary Iodine Activity Concentration	0.1 µCi/gm DE I-131	0.05 µCi/gm DE I-131	Proposed TS change
SGTR Parameters			
Reactor Trip Time (sec)	173.3	No Change	
Safety Injection Signal (sec)	173.3	185	Conservative value based on the Westinghouse T&H analysis
Operator Action to isolate Affected SG (min)	30	55	Conservative value confirmed in Operator timing studies
Action to Align RHRS (hr)	24	29	Conservative assumption that RHRS start is delayed to 29 hrs.
Release to Environment (hr)			
Unaffected SG	0 – 24	0 – 29	Conservative assumption that RHRS start is delayed to 29 hrs.
Affected SG	0 – 0.5	0 – 0.92	Time for operator action to close PORV was increased to 55 minutes
Reactor coolant mass (gm)	1.19E+08	No Change	
Initial Steam Generator Liquid Mass (lbm/SG)	84,000 – Min vol. used to minimize hold-up in the SG	No Change	
	97,064 – Max vol. used to maximize secondary side activity	No Change	

Table 3.4-4 Basic Data and Assumptions for SGTR

Parameter or Assumption	CLB Value	Proposed Value	Reason for Change
Tube Rupture Break Flow (lbm)			The time assumed to close the PORV increased from 30 minutes to 55 minutes. Conservatively, the break flow rate at 30 minutes is assumed to persist for an additional 25 minutes.
Pre-Trip	16,900 {5,850 lbm/min}	No Change	
Post-Trip	138,000 {5,088 lbm/min}	265,200	
Tube Rupture Break Flow Flashing Fraction			
Pre-Trip	0.1993	No Change	
Post Trip	0.1476		
Steam Release (lbm/min)			
Ruptured SG			
Pre-Trip	6.47E+04	No Change	
Post-Trip	3.19E+03	No Change	
Intact SG			
Pre-Trip	6.47E+04	No Change	
Trip – 2 hr	1.99E+03	No Change	
2 – 8 hr	1.36E+03	No Change	RHR cut-in time was increased to 29 hours. The steam release rate from 8 to 24 hours was conservatively maintained for an additional 5 hours.
8 – 24 hr	6.90E+02	No Change	
24 – 29 hr	0	6.90E+02	

Table 3.4-4 Basic Data and Assumptions for SGTR

Parameter or Assumption	CLB Value	Proposed Value	Reason for Change
Release points Ruptured SG Pre-Trip Post-Trip Intact SG Pre and Post-Trip	 Condenser SG Power Operated Relief Valves (PORVs) PORVs	 PORVs PORVs	The current analysis credits the condenser until reactor trip. The revised analysis assumes Loss of Offsite Power (LOOP) coincident with the accident. All releases from the ruptured and intact SG will release through the PORVs.
Operator Action to close Affected SG PORV (min)	30	55	Conservative value confirmed in Operator timing studies
EAB X/Q (sec/m^3) 0 – 2 hr	2.232E-04	1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)

Table 3.4-4 Basic Data and Assumptions for SGTR			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Control Room			
Control Room Isolation (sec)	300	195	Current value was reduced to remove conservatism. Revised Control Room isolation includes SI signal at 185 seconds + 10 seconds for Control Room Damper closure.
Control Room Post Accident Recirculation System (CRPARS) Ventilation (sec)	300	318	CRPARS initiation includes SI signal at 185 seconds + 10 seconds for Diesel start + 63 seconds for Diesel sequencing + 60 seconds for CRPARS damper to open
Control Room Unfiltered Inleakage (cfm)	1000	800	Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20)
Control Room HVAC Parameters (cfm) Unfiltered Inleakage Unfiltered Make-up Air Filtered Recirculation	**0 – 300 s** / **300 s – 30 d** 0 / 1000 2750 / 0 0 / 2250	**0 – 195 s** / **195 – 318 s** / **318 – 30 d** 0 / 800 / 800 2750 / 0 / 0 0 / 0 / 2250	Unfiltered inleakage is not assumed until the control room is isolated at 195 seconds. Inleakage is assumed at 800 cfm, consistent with other DBA analyses.
Control Room Volume (ft^3)	127,600	No Change	
Normal Ventilation Unfiltered Makeup Air (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	

Table 3.4-4 Basic Data and Assumptions for SGTR			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
CRPARS Filter Efficiency (%) Elemental Organic Particulate	 90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	No Change	
Control Room X/Q (sec/m^3)	0 – 8 h 2.93E-3 8 – 24 h 1.73E-3 1 – 4 d 6.74E-4 4 – 30 d 1.93E-4	Affected "B" SG / Intact "A" SG 0 – 0.055 h 7.92E-3 2.24E-3 0.055 – 2 h 5.84E-3 2.46E-3 2 – 8 h 2.34E-2 2.13E-3 8 – 24 h 8.67E-3 8.60E-4 1 – 4 d 6.97E-3 6.96E-4 4 – 30 d 6.41E-3 5.81E-4	NEW ARCON96 X/Q values "B" SG 0 – 2 hour values from Table 1.3-4 have been reduced by a factor of 5 due to plume rise (see sec. 3.4.5.3). "A" SG PORVs have a horizontal exhaust, therefore plume rise reduction for the "A" SG X/Qs cannot be made. Prior to CR isolation (0.055 h) the X/Q is to the CR intake. Post isolation, the X/Q represents the worst CR inleakage pathway into the turbine building.

* The density used to convert volumetric leak rates (gpd) to mass leak rates (lbm/hr) was consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications.

3.4.5.3 SGTR Plume Rise Determination

Following the guidance of RG 1.194, the buoyant plume rise associated with energetic releases from steam relief values or atmospheric steam dumps can be credited if (1) the release is uncapped and vertical, and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed, at the release point height, by a factor of 5.

The 95th percentile wind velocity was determined using meteorological data from 2002-2006. The value of the 95th percentile 10 meter and 60 meter wind speeds was found to be 7.6 and 11.6 meters per second, respectively. The B steam generator PORV has a larger atmospheric dispersion factor than the A PORV because of the close proximity of B PORV to the control room intake and turbine building intake locations. The steam flow from the B PORV is vertical and uncapped at the point where it enters the atmosphere. The elevation at which the steam enters the atmosphere is 682'1" or 23.34 meters above grade. Using linear interpolation, the 95th percentile wind speed at this elevation is 8.6 meters per second. Five times this speed is 43 meters per second.

With a PORV exhaust stack cross sectional area of 2.02 square feet, the flow from an open PORV would need to equal or exceed 632 lbm/min* to equal an exit velocity of 43 meters per second. From Table 3.4-4, the steam flow from the affected steam generator exceeds 632 lbm/min for the entire accident duration. For conservatism, only the 0-2 hour X/Q for the "B" (Affected) SG PORV release is reduced by a factor of five, crediting the plume rise reduction allowed by RG 1.194.

* (conservatively assumed at atmospheric pressure saturated steam conditions)

Table 3.4-5 Time Line of Events

Time, post accident		Event
seconds	hours	
0	0	SGTR – PORV sticks open LOOP
173.3	0.0481	Reactor Trip
185	0.0514	SI Actuated
195	0.0542	Control Room Isolates
318	0.0883	CRPARS initiated
3,300	0.92	PORV Closed (Affected SG Release Terminated)
104,400	29	RHRS Placed In Service (Intact SG Release Terminated)
2,592,000	720	Event Terminated

Table 3.4-6 RCS Break Flow to Affected Steam Generator

Time period (hour)		Total Break Flow Rate	Flashed Break Flow Rate	Liquid Break Flow Rate
From	To	(lbm/min)	(lbm/min)	(lbm/min)
0	0.0481	5850	1166	4684
0.0481	0.92	5088	756	4332
0.92	720	0	0	0

Table 3.4-7 Affected Steam Generator Steam Release to Environment

Time period (sec)		Time period (hour)		Release Rate
From	To	From	To	(lbm/min)
0	173.3	0	0.0481	64,668
173.3	3300	0.0481	0.92	3,186
3300	2,592,000	0.92	720	0

Table 3.4-8 Intact Steam Generator Steam Release to the Environment

Time period, sec		Time period, hour		Release Rate
From	To	From	To	(lbm/min)
0	173.3	0	0.0481	64,668
173.3	7,200	0.0481	2	1,992
7,200	28,800	2	8	1,356
28,800	86,400	8	24	690
86,400	104,400	24	29	690
104,400	2,592,000	29	720	0

3.4.6 SGTR Analysis Results

The results of the analyses are presented in Table 3.4-9 for the Concurrent Spike and for the Pre-accident Iodine Spike.

Table 3.4-9 Dose Summary for the SGTR Accident

Location	TEDE (rem)	Limits (rem)
Concurrent Iodine Spike		
EAB	0.2	2.5
LPZ	0.1	2.5
Control Room	1.1	5
Pre-Accident Iodine Spike		
EAB	0.3	25
LPZ	0.1	25
Control Room	3.9	5

3.5 Main Steam Line Break Analysis

This section describes the methods employed and results of the Main Steam Line Break (MSLB) design basis radiological analysis. This analysis includes doses associated with the releases of radioactive material initially present in primary and secondary liquids at maximum allowable Technical Specification concentrations and adjusting for iodine spiking scenarios. No fuel failure is expected. Doses were calculated at the exclusion area boundary (EAB), at the low population zone (LPZ), and in the Control Room. The methodology used to evaluate the control room and offsite doses resulting from the MSLB accident is consistent with Regulatory Guide 1.183 (Reference 1) in conjunction with TEDE radiological units and limits, ARCON96 based onsite atmospheric dispersion factors, and Federal Guidance Report No. 11 and 12 (References 15 & 16, respectively) dose conversion factors.

3.5.1 MSLB Scenario Description

The Main Steam Line Break (MSLB) accident begins with a break in one of the main steam lines leading from a steam generator (affected generator) to the turbine. Main steam line piping exits the containment and remains interior to the auxiliary building until entering into the turbine building. The control room is within this building matrix, with adjacent walls and entrances to both the auxiliary and turbine buildings. As discussed in Section 3.1.1, the primary pathway and assumed source of inleakage into the control room is through doors adjacent to the turbine building.

In order to determine maximum control room dose, both a steam line break in the turbine building and a break in the auxiliary building were separately evaluated (see Figures 3.5-1 and 3.5-2). This is a change from the current MSLB analysis which assumes the break releases directly into the atmosphere. Each evaluation considered conservative and bounding assumptions to determine which pathway scenario resulted in the maximum control room and offsite dose consequences.

The worst case evaluated MSLB scenario for the EAB, LPZ and control room, involves a steam line break in the turbine building. This scenario will form the conditions, requirements and assumptions for the design basis MSLB accident.

MSLB in the Turbine Building

The affected generator will dry out quickly and release all of the activity initially in the affected generator bulk liquid within 10 minutes directly into the turbine building where there are direct unfiltered inleakage pathways into the control room. Participation with 50% of the turbine building volume is credited for activity entering the building from the break. The pressure surge caused by the steam break will open turbine building blow-outs. During the assumed 10-minute initial release of steam generator (SG) contents, radioactive release from the blow-outs are set equal to the steam flow from the break. Releases from the affected SG will continue after blow down due to primary-to-secondary leakage at the Technical Specification Limiting Condition for Operation 3.4.13.d rate of 150 gallons per day until the MSIV is closed by operator action, which was conservatively assumed to occur at 8 hours. Radioactivity that escapes through blow-outs is modeled with both low and high volume release rates to negate any benefits. Loss of off-site power is assumed. As a result, the condenser is unavailable. Cool down of the primary system is through the release of steam from the intact generator which is also assumed to have a primary-to-secondary leak at the Technical Specification rate of 150 gallons per day. Intact SG steaming will continue until sufficient cooldown at 29 hours allows use of the residual heat removal system (RHRS).

In accordance with RG 1.183, Appendix E, two independent cases are evaluated. Case one assumes a pre-accident iodine spike, while the second case assumes a concurrent iodine spike.

3.5.2 MSLB Source Term Definition

As with the SGTR accident, the analysis of the MSLB accident indicates that no fuel rod failures occur as a result of the transient. Thus, radioactive material releases are determined by assuming the radionuclide concentrations initially present in primary and secondary liquid at maximum Technical Specification limits and iodine spiking.

The Main Steam Line Break analysis uses the primary and secondary liquid source term discussed in Table 3.4-1 and the pre-accident iodine spike source term discussed in Table 3.4-2. The MSLB analysis also assumes a concurrent iodine spike listed below in Table 3.5-1 corresponding to an accident initiated value 500 times the appearance rate. The appearance rate has decreased by a factor of ten from the current license basis values due to the proposed Technical Specification 3.4.16 RCS limit reduction for normal operation (0.1 µCi/gm DE I-131).

Table 3.5-1 Concurrent Iodine Spike MSLB RCS Concentration

Nuclide	Appearance rate for 0.1 µCi/gm DE I-131 Ci/hr	Spike = 500 MSLB Appearance Rate Ci/hr
I-131	1.80E+00	8.98E+02
I-132	4.75E+00	2.37E+03
I-133	3.10E+00	1.55E+03
I-134	1.93E+00	9.63E+02
I-135	2.26E+00	1.13E+03

3.5.3 MSLB Release Transport

The source term resulting from activity in the primary system coolant and from iodine spiking in the primary system is transported to the SGs by the leak-rate limiting condition for operation of 150 gallons per day per SG specified in the Technical Specifications (TS LCO 3.4.13.d).

For the affected generator, the release pathway is assumed to be directly into the turbine building with no credit taken for holdup, partitioning or scrubbing by the SG liquid. Activity released from the break is assumed to participate with 50% of the turbine building volume. From the turbine building, the activity is assumed to leak into the control room as well as pass into the environment through pressure relief blow-outs located around the turbine building. A portion of the activity released through the blow-outs will disperse in the atmosphere and be pulled back into ventilation intakes and louvers and mix with activity residing in the turbine building. The affected generator will release activity into the turbine building until isolated (8 hours) or until the primary side is cooled to 212°F (69.2 hours). The operator action to isolate the affected SG is currently required to be completed within 10 minutes as part of isolating auxiliary feedwater, but is conservatively delayed for 8 hours to increase the dose consequences. In addition, no credit is taken for the steam line isolation signal that would close the affected MSIV based on SI coincident with HI-HI steam flow.

The affected SG transport model utilized for noble gases, iodine and particulates was consistent with Appendix E of Regulatory Guide 1.183. During the first 10 minutes post-trip, the affected SG is assumed to steam dry as a result of the MSLB, releasing all of the nuclides in the secondary coolant that were initially contained in the SG. During the first 8 or 69.2 hours, the primary coolant is also assumed to leak into the affected SG at the rate of 150 gpd with all activity released unmitigated. After 69.2 hours the RCS will have cooled to below 212°F and the release via this pathway terminates. The primary-to-secondary leak rate path is terminated in 8 hours when operator action to isolate the affected SG is credited.

The intact SG is assumed to leak for 29 hours until shutdown cooling is credited for decay heat removal. The primary-to-secondary technical specification leak rate limit of 150 gpd is assumed to maximize the release rate through the SG PORVs. The tube bundles of the intact SG remain covered during the release because of the availability of the Auxiliary Feedwater System. Releases of iodines and particulates are limited due to moisture carryover or partitioning. Releases of noble gases are assumed to occur directly to the environment without any mitigation or holdup.

There are several nuclide transport models associated with the intact SG. Together, they ensure proper accounting of iodine, particulates and noble gas releases. The intact model includes 2 RCS volumes each with the maximum technical specification source term (16.4 μCi/gm DE Xe-133 and 0.1 μCi/gm DE I-131), one for volume noble gas releases and one for iodine and particulate releases. The first volume has a pathway for releases of noble gas activity to the environment at 150 gpd, with 100 percent efficient iodine and particulate filtration. The transport to the environment of noble gases from the primary coolant and from iodine and particulate daughters released from the "filters" occurs without any mitigation or holdup.

The second RCS volume is used to model releases of radionuclides, which are initially in the intact SG liquid and those entering the SG from the primary to secondary leakage flow, as a result of secondary liquid boiling. Due to iodine partitioning and moisture carryover, 1% of the iodine and particulates in the SG bulk liquid are released to the environment at the steaming rate. The effect of partitioning and moisture carryover is modeled by reducing the steam flow rate by a factor of 100 to conserve radionuclides in the intact SG liquid. Radionuclides initially in the steam space do not provide any significant dose contribution and are not considered.

The pre-accident iodine spike is modeled in the same manner as the technical specification coolant activity model previously discussed.

The concurrent iodine spike model is modeled in the same manner as the technical specification coolant activity model but the iodine spike occurs for 8 hours after which the activity remaining in the primary coolant continues to be released for the remainder of the 29 hours.

3.5.4 MSLB Atmospheric Dispersion Factors

3.5.4.1 MSLB Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194

(Reference 6). The MSLB Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS release points:

- "A" SG PORV
- "B" SG PORV

Control room X/Q values for the 'B' SG PORV to the control room intake were selected to model release points applicable to the affected SG for a MSLB in the turbine building. The X/Q is applied to releases from the turbine building blowouts. The control room intake will be isolated within 13 seconds of accident initiation (< 3 seconds for SI and 10 seconds for control room isolation damper closure). Therefore, any leakage into the control room will be from the turbine building (as discussed in Section 3.1.1) which has primary intake points near the South-West corner of the building. The 'B' SG PORV to control room intake X/Q was selected because of its close proximity to the turbine building intake points and because this source-to-receptor combination results in the highest X/Q values. As shown in Figure 3.1-1, the 'B' SG PORV is in close proximity to the turbine building Fan Room louvers. Table 1.3-4 shows that the 'B' SG PORV X/Q values are the highest of any single release point at KPS. Uncorrected for plume rise, the 'B' SG PORV X/Q values are conservative in comparison to an aggregate X/Q that would exist if calculated for multiple blow-outs located at various locations throughout the turbine building.

The 'A' SG PORV X/Q values were used to model the intact SG releases. The 0-2 hour 'A' SG PORV to control room intake X/Q value was used before isolation (13 seconds). After isolation, the 'A' SG PORV to turbine building fan room west louver X/Q values were used.

3.5.4.2 MSLB Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.5.5 MSLB Key Analysis Assumptions and Inputs

3.5.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a MSLB at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

RADTRAD can model a variety of processes that can attenuate and/or transport radionuclides during a MSLB. There are several aspects of the MSLB analysis that require multiple RADTRAD models due to limitations of the code. This is due primarily to treatment of the source terms and because noble gases are released without mitigation, and iodines and particulates are released crediting partitioning and moisture carryover in the intact SG, with no mitigation in the affected SG. The different models include:

- Pre-incident spike – affected SG
- Pre-incident spike – intact SG
- Coincident spike – affected SG
- Coincident spike – intact SG
- Initial RCS TS activity – affected SG
- Initial RCS TS activity – intact SG
- Secondary side bulk liquid activity – affected SG
- Secondary side bulk liquid activity – intact SG

In order to determine maximum control room dose, both a steam line break in the turbine building and a break in the auxiliary building were evaluated (see Figures 3.5-1 and 3.5-2). The intact SG release is through the PORVs to the environment. Each evaluation considered conservative and bounding assumptions to determine which pathway scenario resulted in the maximum control room and offsite dose consequences. The worst case evaluated MSLB scenario for the EAB, LPZ and control room, involves a steam line break in the turbine building. This scenario will form the conditions, requirements and assumptions for the design basis MSLB accident.

The schematic shown in Figure 3.5-1 provides an overall picture of the design basis MSLB involving a break into the turbine building and releases to the environment. Maximum and minimum values were used for secondary side bulk liquid mass. The minimum value is used to reduce holdup for primary to secondary releases in the intact SG and the maximum value is used to maximize secondary side inventory in the affected SG. This is done to maximize dose from primary to secondary side releases.

The evaluation of the break in the turbine building considered conservative and bounding assumptions to model the release from the affected SG into the turbine building volume. Participation with only 50% of the building volume was credited. Blow-out panels are assumed to open to relieve the pressure surge caused by the steam line break. High and low escape rates from the blow-outs (i.e., 10 building-volumes/hr down to 1 building-volume/hr) are modeled to maximize resulting control room and offsite dose consequences. Although multiple blow-out panels would open due to this event creating an aggregate X/Q to the control room, the highest control room X/Q values shown in Table 1.3-4 were used as conservative values to bound any release configuration that would result from a MSLB. The X/Q values used correspond to the "B" SG PORV release point that models a physical separation as close as 12 meters (see Table 3.1-1) between the PORV to the control room intake and turbine building louvers.

Evaluations for both the pre-accident and the concurrent iodine spike source terms were performed for a MSLB in the turbine building. Based on releases from the affected SG, which will persist for 69.2 hours (the time necessary to cool the primary system down to 212°F), maximum dose consequences were determined for the EAB, LPZ and the control room. It became apparent that control room doses could not be maintained below 5 Rem for the concurrent iodine spike case if the affected generator releases primary coolant unmitigated into the turbine building at 150 gpd for the entire 69.2-hour cooldown period. Operator action is needed to isolate the affected SG within 8 hours to maintain control room dose consequences within allowed limits. Existing Operation procedure steps (Reference 32) have the Operator closing the affected SG MSIV following a MSLB much earlier than 8 hours (< 10 minutes) in order to isolate feedwater flow. For evaluation purposes, an assumption to isolate the affected SG by closing the MSIV within 8 hours was chosen to maximize consequences. This assumption also greatly relaxes the timing and burden on the operator to complete this action.

Steaming from the intact SG continues for 29 hours until RCS pressure reduces to a level where RHRS can be used to remove decay heat.

In accordance with RG 1.183, Appendix E, two independent cases are evaluated. Case one assumes a pre-accident iodine spike, while the second case assumes a concurrent iodine spike. As previously discussed, the concurrent iodine spike case credits operator action to close the affected steam generator MSIV within 8 hours accomplished through existing procedure actions (Reference 32). Conservatively, no action to close the MSIV was assumed for the pre-accident case to maximize dose consequences.

3.5.5.2 Basic Data & Assumptions for MSLB

The Basic Data and Assumptions are listed in Table 3.5-2. Control room information is available in both Tables 3.5-2 and 1.3-1.

Figure 3.5-1 MSLB Radioactive Release into the Turbine Building Schematic (Design Model)

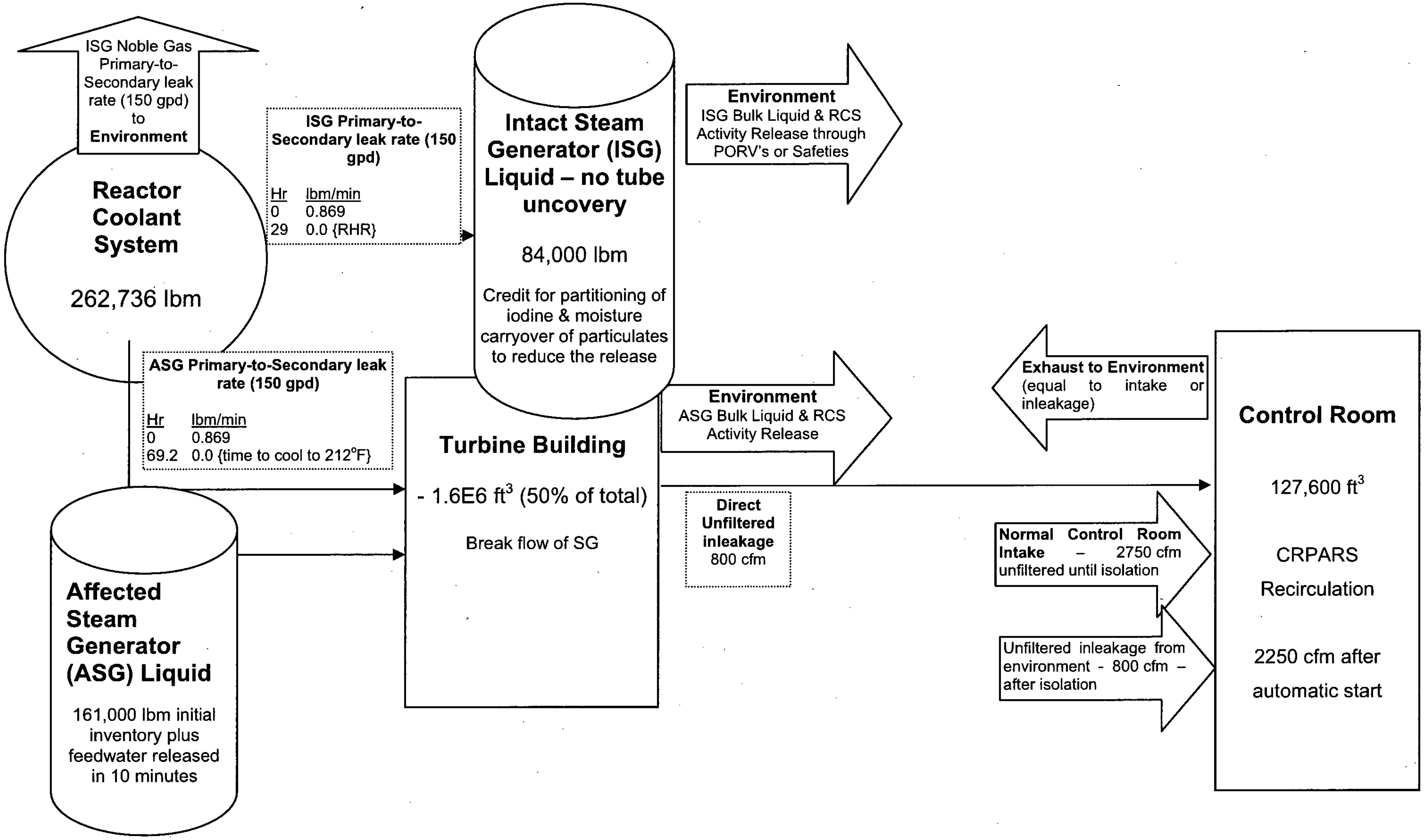


Figure 3.5-2 MSLB Radioactive Release into the Auxiliary Building Schematic

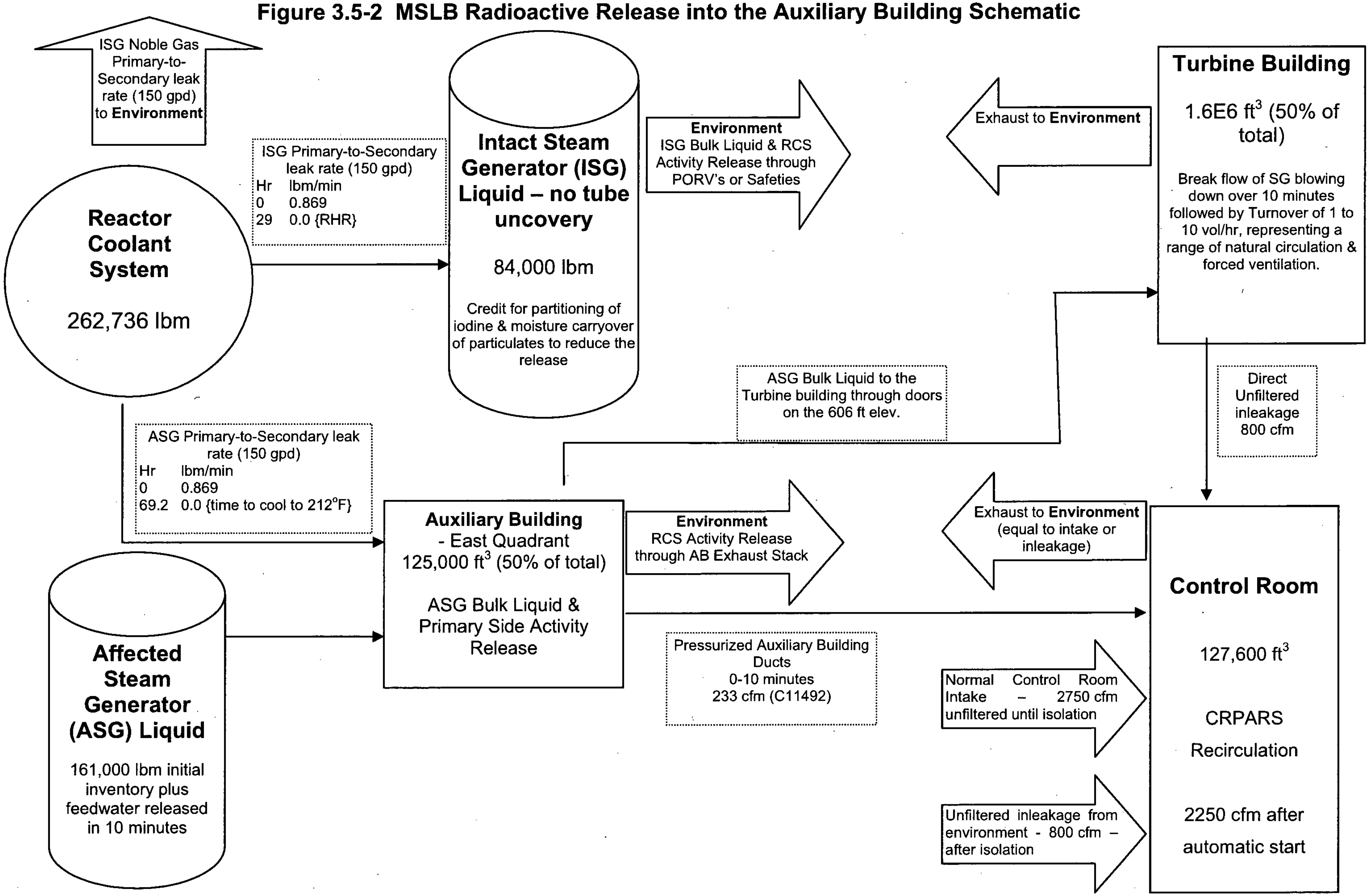


Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
		Source Term	
Primary Coolant Specific Activity Limit			Proposed Technical Specification limit change.
DE I-131 (μCi/gm)	1	0.1	
Gross Activity	Not included	α 0.1 μCi/gm DE I-131	Derived from the 1% failed fuel inventory and equivalent to the failed fuel for the TS DE I-131 limit.
Primary Coolant Concentrations at TS Limit μCi/gm I-131 I-132 I-133 I-134 I-135	 7.80E-01 7.93E-01 1.16E+00 1.61E-01 6.37E-01	 7.82E-02 7.97E-02 1.17E-01 1.62E-02 6.40E-02	Current values include 5% variation to consider minor variations in fuel design (e.g., enrichment, core mass and cycle length). Proposed values are adjusted to allow 10% variation, to make consistent with similar allowance built into core inventory curies.
Primary Coolant Noble Gas Activity	1% fuel defects	16.4 μCi/gm DE Xe-133	The revised Noble gas limit corresponds to an equivalent level of fuel failure (0.027%) as the TS DE I-131 limit of 0.1 μCi/gm
Accident Initiated (Concurrent) Iodine Spike	500	No Change	
Accident-Initiated (Concurrent) Spike Duration (hr)	4	8	Per RG 1.183

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Iodine Appearance Rate I-131 I-132 I-133 I-134 I-135	<u>Ci/min</u> 0.301 0.788 0.519 0.319 0.377	<u>Ci/hr</u> 1.80 4.75 3.10 1.93 2.26	The difference in the iodine appearance rates reflects a unit conversion and a factor of ten reduction directly related to the reduced TS specific activity limit. Values on a consistent unit basis are shown to the right. <u>Ci/min</u> 0.030 0.079 0.052 0.032 0.038
Primary to Secondary Leak rate (gpd/SG)*	150	No Change	
Pre-Accident Spike Coolant Activity (μCi/gm DE I-131)	60	10	Proposed TS spike limit was lowered commensurate with primary coolant activity reduction.
Iodine Partitioning in Intact SG	PC for iodine = 100	No Change	
Iodine chemical form of Primary-to-Secondary Leakage (%)	Elemental 97 Organic 3 Particulate 0	No Change	
Moisture Carryover in Intact SG	1%	No Change	

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
SG Iodine Partition Factor Faulted SG Intact SG	 1.0 0.01	No Change	
Secondary Iodine Activity Concentration	0.1 μCi/gm DE I-131	0.05 μCi/gm DE I-131	Proposed TS change
MSLB Parameters			
Safety Injection Signal (sec)	0	<3	This time is based on high-high steam flow signal on the intact SG of 2.9 seconds. The SI signal actually comes in based on a low-low steam pressure on the affected SG <<3 seconds.
Operator Action to close Affected SG MSIV (hr)	NA	8	The concurrent iodine spike in the turbine building analysis requires closing the MSIV on the affected SG within 8 hours to maintain resulting control room dose within GDC 19 limits. Current analysis does not assume affect SG isolation
Action to Align RHRS (hr)	24	29	Conservative assumption that RHRS start is delayed to 29 hrs.
Reactor coolant Mass (gm)	1.19E+08 (262,736 lbm)	No Change	

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Release to Environment (hr)			
Unaffected SG	0 – 24	0 – 29	Conservative assumption that RHRS start is delayed to 29 hrs.
Affected SG			
Pre-accident spike	72	69.2	The current basis T&H analysis is cooldown to 212°F in 69.2 hr. Use of 69.2 hr is a reduction in conservatism.
Concurrent spike	72	8	Operator Action credited for the concurrent spike - utilizing existing procedure actions to isolate the affected SG.
Release of Initial Mass in Faulted Generator (min)	2	10	Validation of Operator actions shows isolation of feedwater to the affected SG will take up to 10 minutes. Extending the release duration to 10 minutes ensures that all of the Curies are released from the affected SG and results in higher doses.
Faulted SG Steam Mass (lbm)	4759	No Change	

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Initial SG Liquid Mass (lbm)			Total Faulted SG mass = 161,000 lbm (Steam and Liquid mass)
Faulted SG	156,254	No Change	
Intact SG	84,000 – Min volume used to minimize hold-up in the SG	No Change	
Faulted SG Release (lbm)			The initial mass from the faulted SG is released over 2 minutes.
0 – 2 min	1.61E+05	No Change	
2 – 10 min	0	1.03E+05 (feedwater)	Extending the period of release assures all activity that initially was in the SG is released.
10 – 30 min	0	0	
Intact SG Release (lbm)			RHR cut-in time was increased to 29 hours. The steam release rate from 8 to 24 hours was conservatively maintained for an additional 5 hours.
0 – 2 hr	2.22E+05	No Change	
2 – 8 hr	4.24E+05	No Change	
8 – 24 hr	6.14E+05	No Change	
24 – 29 hr	0	1.92E+05	
Turbine Building Volume (ft^3)	NA	3.19E+06	50% credit = 1.60E+06 ft^3 Worst case MSLB occurs in the turbine building. Current basis assumes MSLB occurs into the environment.

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Release points MSL Break (Affected SG)	Environment	'B' SG is the affected generator releasing into the turbine building and released to the environment from blowout panels.	The current analysis assumes a break directly into the environment since the method utilized only one station control room X/Q value that was supposed to represent and bound all possible release points. A break into the turbine building is bounding over a break into the auxiliary building.
Intact SG	Environment	'A' SG PORV	To maximize control room dose, the 'B' SG is modeled as the affected SG. Therefore, the 'A' SG PORV represents the intact SG.
Turbine Building Release to Environment (cfm)	NA	0 – 2 min 2.16E+06 2 – 10 min 3.45E+05 10 min – 30 d 2.67E+04	Affected SG blow down occurs into the turbine building (TB) in the first 2 minutes. From 2 to 10 minutes, until feedwater is isolated, steaming continues. After 10 minutes, primary to secondary releases from the affected SG continue into the TB. One TB volume turnover per hour was assumed to maximize control room dose. Evaluation up to 10 volumes/hour (2.67E+05 cfm) showed decreasing control room doses and relative insensitivity of the offsite results to large changes in volumetric release rates.

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
EAB X/Q (sec/m^3) 0 – 2 hr	2.232E-04	1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
Control Room			
Control Room Isolation (sec)	300	13	Current value was reduced to remove conservatism. Revised Control Room isolation includes SI signal at 3 seconds + 10 seconds for Control Room Damper closure.
Control Room Post Accident Recirculation System (CRPARS) Ventilation (sec)	300	136	CRPARS initiation includes SI signal at 3 seconds + 10 seconds for Diesel start + 63 seconds for Diesel sequencing + 60 seconds for CRPARS damper to open.
Control Room Unfiltered Inleakage (cfm)	1000	800	Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20).

Table 3.5-2 Basic Data and Assumptions for MSLB

Parameter or Assumption	CLB Value	Proposed Value	Reason for Change
Control Room HVAC Parameters (cfm) Unfiltered Inleakage Unfiltered Make-up Air Filtered Recirculation	<u>0 – 300 s</u> / <u>300 s – 30 d</u> 0 / 1000 2750 / 0 0 / 2250	<u>0 – 13s</u> / <u>13 – 136s</u> / <u>136s – 30 d</u> 0 / 800 / 800 2750 / 0 / 0 0 / 0 / 2250	Unfiltered inleakage is not assumed until the control room is isolated at 13 seconds. Inleakage is assumed at 800 cfm, consistent with other DBA analyses.
Control Room Volume (ft^3)	127,600	No Change	
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
CRPARS Filter Efficiency (%) Elemental Organic Particulate	 90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	No Change	
Control Room X/Q (sec/m^3)	0 – 8 h 2.93E-3	<u>ASG</u> / <u>ISG</u> 0 – 2 h 3.96E-2 / 2.46E-3 2 – 8 h 3.20E-2 / 2.13E-3 8 – 24 h 1.21E-2 / 8.60E-4	NEW ARCON96 X/Q values Affected SG (ASG) utilizes X/Q values from Table 1.3-4 for the 'B' SG PORV to the control room

Table 3.5-2 Basic Data and Assumptions for MSLB			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
	8 – 24 h 1.73E-3 1 – 4 d 6.74E-4 4 – 30 d 1.93E-4	1 – 4 d 1.01E-4 6.96E-4 4 – 30 d 8.58E-3 5.81E-4	intake to maximize CR dose. With only a 12 meter separation from the 'B' PORV to the intake, the high X/Q bounds any aggregate X/Q that would result from modeling TB blowout panels to the nearest intake point. Intact 'unaffected' SG (ISG) releases from 'A' PORV to the TB west louvers maximize the CR X/Q from the 'A' SG. For the first 13 seconds, prior to CR isolation, the X/Q to the normal control room intake is 2.24E-3.

* The density used to convert volumetric leak rates (gpd) to mass leak rates (lbm/hr) was consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications.

3.5.6 MSLB Analysis Results

The total TEDE to the EAB, LPZ and Control Room from a Main Steam Line Break is summarized below in Table 3.5-3 for the concurrent and pre-accident spike. The concurrent spike results in the highest dose consequences for both offsite and the control room. All doses are within the limits specified in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 3.5-3 Dose Summary for the MSLB Accident

Location	TEDE (rem)	Limits (rem)
Concurrent Iodine Spike		
EAB	0.1	2.5
LPZ	0.1	2.5
Control Room	4.2	5
Pre-Accident Iodine Spike		
EAB	0.1	25
LPZ	0.1	25
Control Room	4.7	5

3.6 Locked Rotor Accident (LRA) Analysis

This section describes the methods employed and results of the Locked Rotor Accident (LRA) design basis radiological analysis. The analysis assumes failure of 25% of the fuel rods, due to Departure from Nucleate Boiling (DNB) during the accident. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the KPS control room. The methods used to evaluate the control room and offsite doses resulting from the LRA included Regulatory Guide 1.183 methodology, ARCON96-based control room atmospheric dispersion factors, PAVAND-based EAB and LPZ atmospheric dispersion factors, Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors, and credit for a new operator action to actuate the control room emergency ventilation system within one hour of the accident.

3.6.1 LRA Scenario Description

The Locked Rotor Accident (LRA) begins with instantaneous seizure of a rotor in one of the two reactor coolant pumps. The sudden decrease in core coolant flow while the reactor is at power results in a degradation of core heat transfer that results in assumed fuel damage due to Departure from Nucleate Boiling (DNB). Although there is no increase in the leak rate of primary coolant to the secondary side during the LRA, a large amount of activity (from the failed fuel) is transported to the secondary side via any pre-existing leaks in the steam generators.

A turbine trip and coincident loss of offsite power are incorporated into the analysis. This results in a release to the environment via power operated relief valves (PORV) with releases to the environment continuing until cooldown can be performed 8 hours post-accident using the Residual Heat Removal System (RHR). Operator action is credited for control room isolation and emergency ventilation actuation one hour following event initiation.

Kewaunee station is removing credit for the Control Room Ventilation Intake radiation monitor R-23 to provide control room isolation. The R-23 system is not safety grade and consists of a single radiation monitor. In addition, the isolation signal generated by R-23 is only a partial signal that will not assure the closure of all control room inlet and outlet ventilation dampers to provide complete control room isolation. Full isolation requires actions by the operator to close dampers that are not included in the isolation logic. The current Locked Rotor Accident (LRA) uses and credits the R-23 system for control room isolation. The basis behind the use of R-23 relies on arguments that Operations will take appropriate actions within 45 minutes to isolate the control room if R-23 fails to perform its isolation function. Removing credit for R-23 requires an alternative means to ensure control room isolation. Operator action will be required within one hour following a LRA to isolate the control room. One hour is sufficient time for the operator to identify the accident, take necessary emergency steps in response to the accident, and direct action to isolate the control room and start the control room emergency ventilation system. This new time-critical operator action will be incorporated into Operation procedures and validated.

3.6.2 LRA Source Term Definition

The core source term used in the Locked Rotor Analysis is taken from Table 3.2-3. Analyses are based on 25% of the gap activity being released, with gap activity based on Regulatory Position 3 of RG 1.183.

3.6.3 LRA Release Transport

The release scenario uses the Technical Specification LCO 3.4.13.d primary to secondary leakage limit of 150 gpd per steam generator. The release from both steam generators continues for 8 hours until shutdown cooling can be placed into service to remove decay heat. After 8 hours, the release from the steam generators is terminated.

The RADTRAD-NAI computer code (Reference 3) is used to model the time dependent transport of radionuclides, from the primary to secondary side and out to the environment via steam relief valves.

3.6.4 LRA Atmospheric Dispersion Factors

3.6.4.1 LRA Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The LRA Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source points:

- "A" Steam Generator PORV
- "B" Steam Generator PORV

The control room X/Qs determined represent the highest values calculated based on the shortest distance measured from each applicable source location to control room receptor location (see Figure 3.1-1).

3.6.4.2 LRA Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.6.5 LRA Analysis Assumptions and Key Parameters

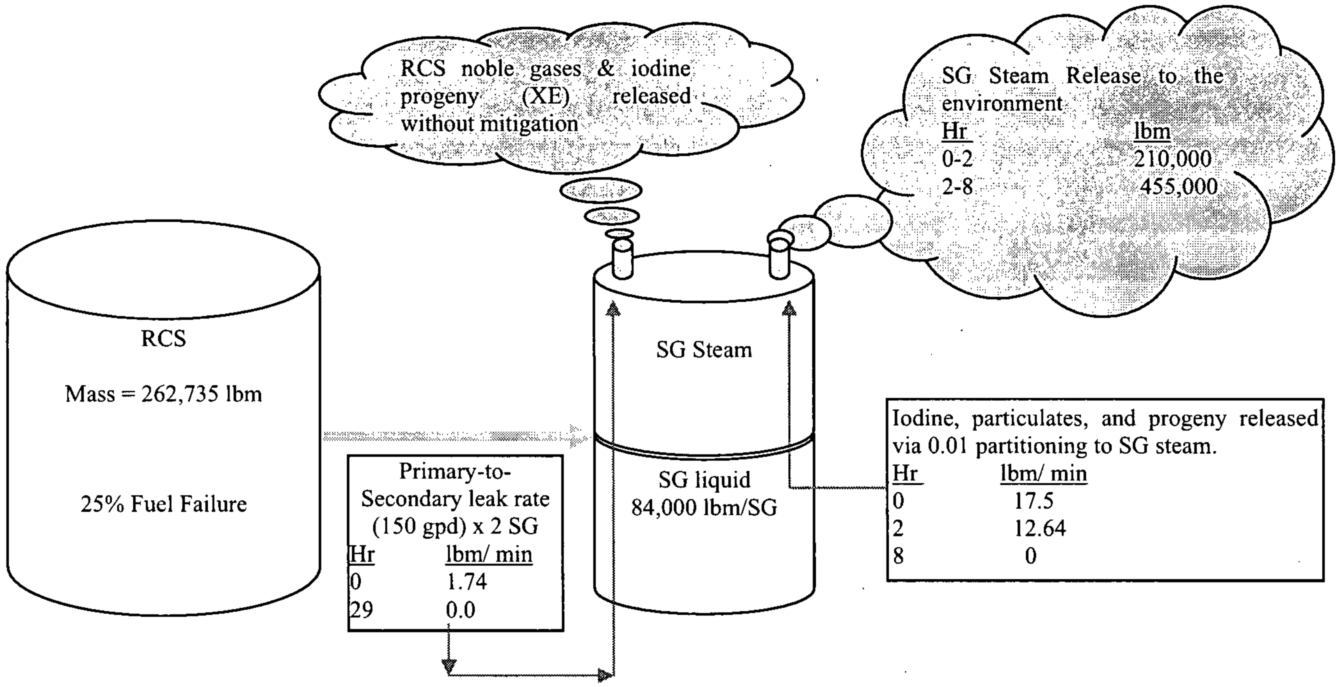
3.6.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a LRA at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

RADTRAD can model a variety of processes that can attenuate and/or transport radionuclides during a LRA. There are aspects of the LRA analysis that require two RADTRAD models due to limitations of the code. This is due primarily to treatment of the source terms because noble gases are released without mitigation and iodines and particulates are released crediting partitioning and moisture carryover. For conservatism, the postulated releases from assumed primary-to-secondary leakage of 150 gpd in each steam generator are combined and released from the generator PORV showing the highest control room X/Q value. The worst case release path for pre and post control room isolation is the "B" Steam Generator PORV. Note that the X/Q values for this pathway were reduced by a factor of 5 for the 0-2 hr and 2-8 hr periods. This reduction is taken following the guidance of RG 1.194, crediting the effects of plume rise for high velocity exhaust steam that exceeds the 95th percentile wind speed by a factor of 5 and adjusted for the physical release elevation. For explanation of this determination, see section 3.6.5.3.

A schematic shown in Figure 3.6-1 provides a summary of the LRA releases to environment.

Figure 3.6-1 LRA Radioactive Release Schematic



3.6.5.2 Basic Data & Assumptions for LRA

Changes have been made to the AST LRA. Table 3.6-1 provides a complete list of inputs and assumptions used to reanalyze the KPS LRA.

Table 3.6-1 Basic Data and Assumptions for LRA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
Primary to Secondary Leak rate (gpd/SG)*	150	No Change	
Failed Fuel Following the Accident (%)	50	25	Rods-in-DNB analysis show approximately 7% rods-in-DNB following a LRA for the current cycle. 25% is specified in the reload safety analysis checklist (RSAC).
Fraction of Core Activity in Gap (%) I-131 Kr-85 Other Noble Gases Other Halogens Alkali Metals	 8 10 5 5 12	 No Change	
Iodine Partitioning	PC = 100	No Change	
Alkali Metal Partitioning	PC = 100	No Change	
Iodine chemical form of Primary-to-Secondary Leakage (%)	Elemental 97 Organic 3 Particulate 0	No Change	
Initial Secondary Side Coolant Activity	Included	Not Included	Because fuel failure occurs, modeling of initial coolant activity is

Table 3.6-1 Basic Data and Assumptions for LRA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
			not required. CLB analysis shows less than 1% contribution to dose from secondary side activity.
Core Activity	Table 3.2-3	No Change	
Radial Peaking Factor	1.7	No Change	
Tube Uncovery.	No tube bundle uncovery assumed.	No Change	
LRA Parameters			
RHR Cut-In Time (hr)	8	No Change	
Reactor Trip Time (sec)	0	No Change	
Loss of Offsite Power (sec)	0	No Change	
Safety Injection Signal	None	No Change	
Reactor coolant mass (gm)	1.19E+08	No Change	
Steam Generator Liquid Mass (lbm/SG) 0 – 30 minutes 30 min – 8 hours	 87,000 116,900	 84,000 No Change	Minimum SG liquid volume used to minimize hold-up

Table 3.6-1 Basic Data and Assumptions for LRA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Steam Release (lbm) 0 – 2 hr 2 – 8 hr	 210,000 455,000	No Change	
Release point	Not applicable (One site control room X/Q represented any release point to the control room)	'B' Steam Generator PORV	New ARCON96 estimates of control room X/Q (Table 1.3-4) show the 'B' PORV has the highest dispersion factor to the control room of any applicable release pathway.
EAB X/Q (sec/m^3) 0 – 2 hr	2.232E-04	1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)

Table 3.6-1 Basic Data and Assumptions for LRA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
		Control Room	
Control Room Volume (ft^3)	127,600	No Change	
Control Room Isolation (min)	10.67	60	CLB credits control room intake radiation monitor R-23 to detect and isolate the control room. R-23 is not redundant and is no longer credited for CR isolation. NEW Operator action is proposed to isolate the control room within 60 minutes of LRA utilizing multiple inputs as indicators of the accident (e.g., Rx coolant low flow and radiation monitor alarms).
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
Control Room Post Accident Recirculation system (CRPARS) Ventilation (min)	11	60	CRPARS initiation is assumed to occur at 60 minutes, coincident with the operator action to isolate the control room.
Control Room Unfiltered Inleakage (cfm)	1500	800	Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20)

Table 3.6-1 Basic Data and Assumptions for LRA					
Parameter or Assumption	**CLB Value**		**Proposed Value**		**Reason for Change**
Control Room HVAC Parameters (cfm)	**0 – 11 m**	**11 m – 30 d**	**0 – 60 m**	**60 m – 30 d**	Unfiltered inleakage is not assumed until the control room is isolated at 60 minutes at which time inleakage is assumed at 800 cfm, consistent with other DBA analyses.
Unfiltered Inleakage	0	1500	0	800	
Unfiltered Make-up Air	2750	0	2750	0	
Filtered Recirculation	0	2250	0	2250	
CRPARS Filter Efficiency (%)					
Elemental	90 (includes safety factor of 2)		No Change		
Organic	90 (includes safety factor of 2)				
Particulate	99				
Control Room X/Q (sec/m^3)	0 – 8 h	2.93E-3	0 – 2 h	7.92E-3	0 – 8 hour values from Table 1.3-4 have been reduced by a factor of 5 due to plume rise (see Section 3.6.5.3).
			2 – 8 h	6.40E-3	
	8 – 24 h	1.73E-3		1.21E-2	
	1 – 4 d	6.74E-4		1.01E-2	Prior to CR isolation the X/Q is to the CR intake. Post isolation, the X/Q represents the worst CR inleakage pathway into the turbine building.
	4 – 30 d	1.93E-4		8.58E-3	

* The density used to convert volumetric leak rates (gpd) to mass leak rates (lbm/hr) was consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications.

3.6.5.3 Plume Rise Determination

Following the guidance of RG 1.194, the buoyant plume rise associated with energetic releases from steam relief valves or atmospheric steam dumps can be credited if (1) the release is uncapped and vertical, and (2) the time-dependent vertical velocity exceeds the 95th percentile wind speed, at the release point height, by a factor of 5.

The 95th percentile wind velocity was determined using meteorological data from 2002-2006. The value of the 95th percentile 10 meter and 60 meter wind speeds was found to be 7.6 and 11.6 meters per second, respectively. The B steam generator PORV has a larger atmospheric dispersion factor than the A PORV because of the close proximity to the control room intake and turbine building intake locations. The steam flow from the B PORV is vertical and uncapped at the point where it enters the atmosphere. The elevation at which the steam enters the atmosphere is 682'1" or 23.34 meters above grade. Using linear interpolation the 95th percentile wind speed at this elevation is 8.6 meters per second. Five times this speed is 43 meters per second.

With a PORV exhaust stack cross sectional area of 2.02 square feet, the flow from an open PORV would need to equal or exceed 632 lbm/min* to equal an exit velocity of 43 meters per second. The steam flows for the LRA are 210,000 lbm for the first 2 hours and 455,000 lbm for 2 hours to 8 hours. Because the steam release from the LRA is assumed from both generators, the single generator flow rates would be one half or 875 lbm/min for the first 2 hours and 632 lbm/min from 2 to 8 hours following the LRA. For both the 0-2 hour and 2-8 hour periods following the LRA, flow out the PORV is sufficient to achieve an exhaust exit velocity that is five times higher than the 95th percentile wind speed. Therefore, the calculated ARCON96 control room X/Q values from the B PORV were reduced by a factor of 5 for the 0-2 and 2-8 hour periods.

* (conservatively assumed at atmospheric pressure saturated steam conditions)

3.6.6 LRA Results

The results of the design basis Locked Rotor analysis are presented in Table 3.6-2. These results show the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30-day duration of the event for the control room and the LPZ. The doses are calculated with the TEDE methodology, and are compared with the applicable acceptance criteria specified in 10 CFR 50.67 and Regulatory Guide 1.183.

Table 3.6-2 TEDE Results for the Locked Rotor Accident

Location	TEDE (rem)	Limits (rem)
EAB	0.3	2.5
LPZ	0.2	2.5
Control Room	4.7	5

3.7 RCCA Ejection Accident (REA) Analysis

This section describes the evaluation of TEDE at the EAB, LPZ and Control Room from a KPS Rod Control Cluster Assembly (RCCA) Ejection Accident (REA). Two release cases are considered. The first case is a release into the containment. The second case is a release into the primary coolant, which is subsequently released through the secondary system.

3.7.1 REA Scenario Description

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion in conjunction with an adverse core power distribution, possibly leading to localized fuel rod damage.

3.7.2 REA Source Term Definition

The core source term used in the RCCA Ejection Accident Analysis are taken from Table 3.2-3. The release of the core source term is adjusted for the fraction of fuel rods assumed to fail during the accident and the fractions of core inventory assumed to be in the pellet-to-clad gap.

Less than 15 percent of the fuel rods in the core undergo DNB as a result of the rod-ejection accident. In determining the offsite doses following a rod-ejection accident, it is conservatively assumed that 15 percent of the fuel rods in the core suffer sufficient damage such that all of their gap activity is released. Ten percent of the total core activity of iodine and noble gases, and 12 percent of the total core activity for alkali metals are assumed to be in the fuel-cladding gap. In the calculation of activity releases from the failed/melted fuel, the maximum radial peaking factor of 1.7 was applied.

A small fraction of the fuel in the failed fuel rods is assumed to melt as a result of the rod ejection accident. This amounts to 0.375 percent of the core, and the melting takes place in the centerline of the affected rods. The 0.375 percent of the fuel assumes that

15 percent of the rods in the core enter DNB. Of the rods that enter DNB, 50 percent are assumed to experience some melting of the fuel (7.5 percent of the core). Of the rods experiencing melting, 50 percent of the axial length of the rod is assumed to experience melting (3.75 percent of the core). It is further assumed that only 10 percent of the radial portion of the rod experiences melting (0.375 percent of the total core).

For both the containment leakage release path and the primary-to-secondary leakage release path, all noble gas and alkali metal activity released from the failed fuel (both gap activity and melted fuel activity) is available for release.

For the containment leakage release path, all of the iodine released from the gap of failed fuel and 25 percent of the activity released from melted fuel is available for release from containment.

The release fractions for both the containment and secondary system release scenarios were calculated as follows, using the design input and assumptions provided in Table 3.7-2.

Input	Description	Value
A	Radial Peaking Factor	1.7
B	Rods > DNB	15.0%
C	% Rods > DNB with centerline melt	50.0%
D	% inner rod melt limit	10.0%
E	% axial length with melt	50.0%
F	Cesium Gap Fraction	12.0%
G	Iodine and Noble Gas Gap Fractions	10.0%
H	Fuel Noble Gas Available for Release	100.0%
I	Fuel Iodine Available for Release to Containment	25.0%
J	Fuel Iodine Available for Release to Secondary System	50.0%
K	Cesium in Fuel Available for Release	100.0%

Percent of core fuel volume that is melted:

$$L = B * C * D * E = 0.375\%$$

Percent of iodine core inventory released in the containment release scenario:

$$((G * B) + (I * L)) * A = 2.71\%$$

Percent of iodine core inventory released in the secondary side release scenario:

$$((G * B) + (J * L)) * A = 2.87\%$$

Noble gas release fraction used for both scenarios:

$$((G * B) + (H * L)) * A = 3.19\%$$

Cesium release fraction used for both scenarios:

$$((F * B) + (K * L)) * A = 3.70\%$$

3.7.3 REA Release Transport

Two release paths are considered for the REA: containment leakage and the secondary system.

The containment release transport assumptions and methodology are similar to the LOCA and can be found in section 3.2.5, with a few exceptions. The exceptions are:

1) The core release fractions are based on Appendix H of R.G. 1.183. The core release fractions are based on the consequences of 15% failed fuel and 0.375% melted fuel.
2) Containment sprays do not initiate due to a REA. Therefore there are no consequences from ECCS leakage and RWST back-leakage.
3) The safety injection signal is initiated 4 minutes after a REA. Therefore, the control room is not isolated until 4 minutes 10 seconds following a REA.

The second release path is via the secondary system. The activity in the secondary system release is based on Appendix H of RG 1.183. The iodines released from the steam generators are assumed to be 97% elemental and 3% organic. The maximum allowable primary-to-secondary leak rate of 150 gpd per steam generator, which is specified in Technical Specification LCO 3.4.13.d, exists until shutdown cooling is in operation and release from the steam generators terminate. All noble gas radionuclides released to the secondary system are released to the environment without reduction or mitigation. The condenser is not available due to an assumed loss of offsite power. A partition coefficient for iodine of 100 is assumed in the steam generators.

The primary-to-secondary leak occurs during the first 30 minutes of the REA (until primary system pressure is less than secondary side system pressure). Steam generator mass releases are unchanged from previous Westinghouse thermal-hydraulic analyses. The steam released during the REA and subsequent cool-down is listed in Table 3.7-2.

3.7.4 REA Atmospheric Dispersion Factors

3.7.4.1 REA Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The REA Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source points:

- Reactor Building Exhaust Stack
- Shield Building
- Auxiliary Building Exhaust Stack
- "A" Steam Generator PORV
- "B" Steam Generator PORV

The control room X/Qs determined represent the highest values calculated based on the shortest distance measured from each applicable source location to control room receptor location (see Figure 3.1-1).

3.7.4.2 REA Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.7.5 REA Analysis Assumptions and Key Parameters

3.7.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from the containment airborne release and primary-to-secondary release resulting from a REA at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

Table 3.7-1 shows the timing of events used in the REA analysis. The sequencing of events is derived from the design inputs and assumptions listed in Table 3.7-2.

Table 3.7-1 REA Event Timing

			Event Timing	
Statepoint	**Description**	**Relationship**	**(min)**	**(hr)**
T0	Start of Event (instantaneous release)	0 sec	0	0
T1	SI Signal	T0 + 240 sec	4	0.06667
T2	Control Room Isolation	T1 + 10 sec	4.1667	0.06944
T3	CRPARS Starts	T1 + 133 sec	6.2167	0.10361
T4	Shield Bldg Ventilation Starts	T1 + 10 min	14	0.23333
T5	Secondary Side Releases Terminate	T0 + 0.5 hr	30	0.5
T6	Shield Bldg Recirculation Starts	T1 + 0.5 hr	34	0.56667
T7	Containment Leak Rates Decrease by 50%	T0 + 24 hr	1440	24
T8	End of Event	T0 + 720 hr	43200	720

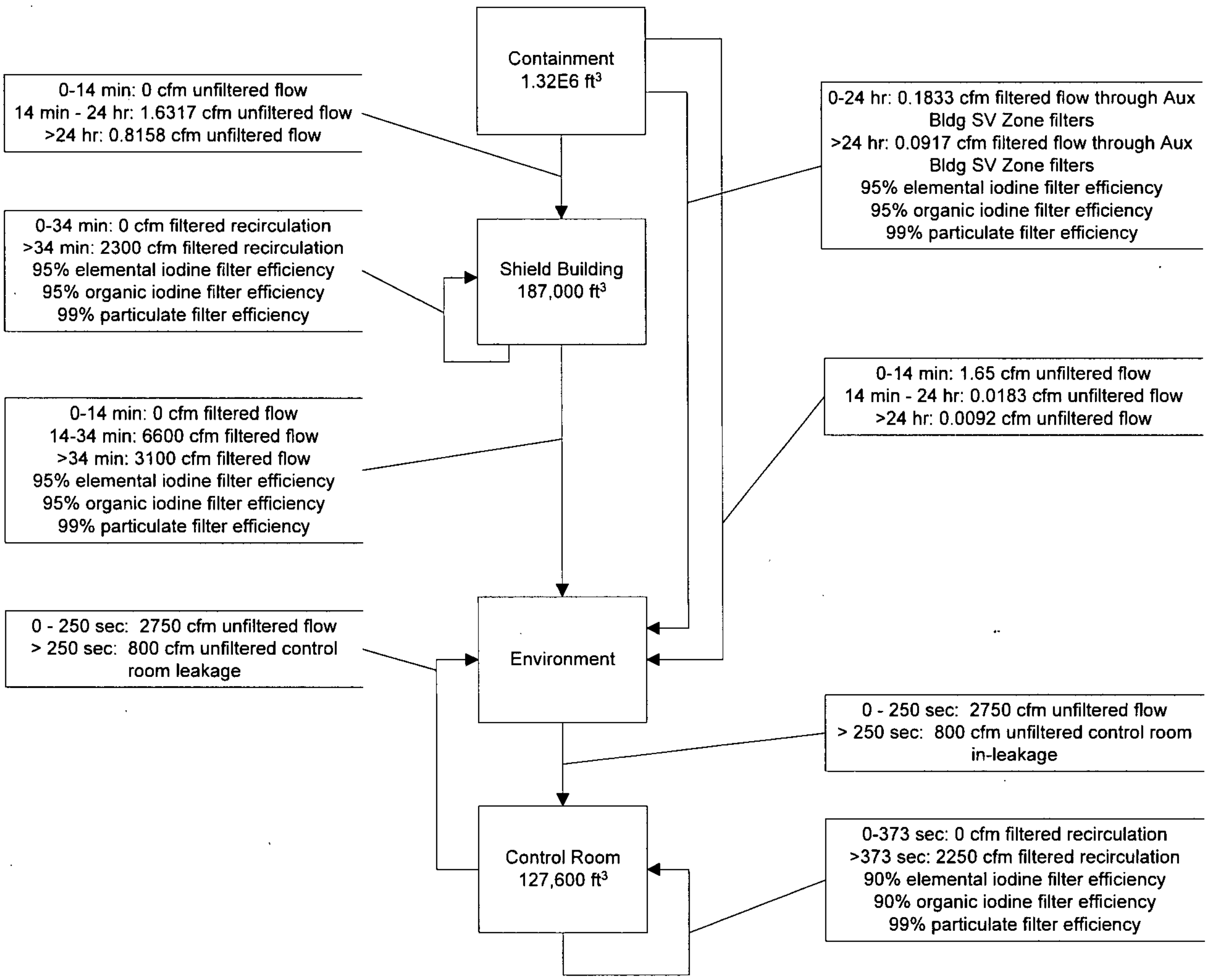
A schematic shown in Figure 3.7-1 provides a flowchart demonstrating the compartments and pathways used in RADTRAD to calculate the doses resulting from containment releases. Figure 3.7-2 provides a similar flowchart for secondary system releases resulting from a REA.

Containment Leakage Model

The primary containment leakage model assumes the failed fuel enters the containment and is released to the atmosphere through containment leakage. The natural deposition mechanism within the containment volume is modeled using the Powers model built into RADTRAD. Containment spray removal is not credited. The containment leaks directly to the environment, through the shield building, and through the auxiliary building special ventilation zone. Releases from the auxiliary building special ventilation zone to the environment are filtered. The shield building ventilation system filters the shield building air volume. A portion of the shield building air volume is discharged to the environment as necessary to maintain the negative pressure in the shield building annulus. Releases from the shield building to the environment are filtered. The shield building ventilation system fans establish a negative building pressure within the first 10 minutes after the safety injection signal. During that interval no credit is taken for filtering the shield building exhaust.

During the first 10 minutes of the accident, it is assumed that 90 percent of the activity leaking from the containment is discharged directly to the environment and 10 percent enters the Auxiliary Building where it is released through filters. After 10 minutes, only 1.0 percent of the activity leaking from the containment is assumed to go directly to the environment, 10 percent continues to go to the Auxiliary Building, and 89 percent is assumed to pass into the Shield Building. The air discharged from the Shield Building is filtered to remove iodine. Additionally, once the Shield Building is brought to subatmospheric pressure at 30 minutes into the event, the iodine is subject to removal by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

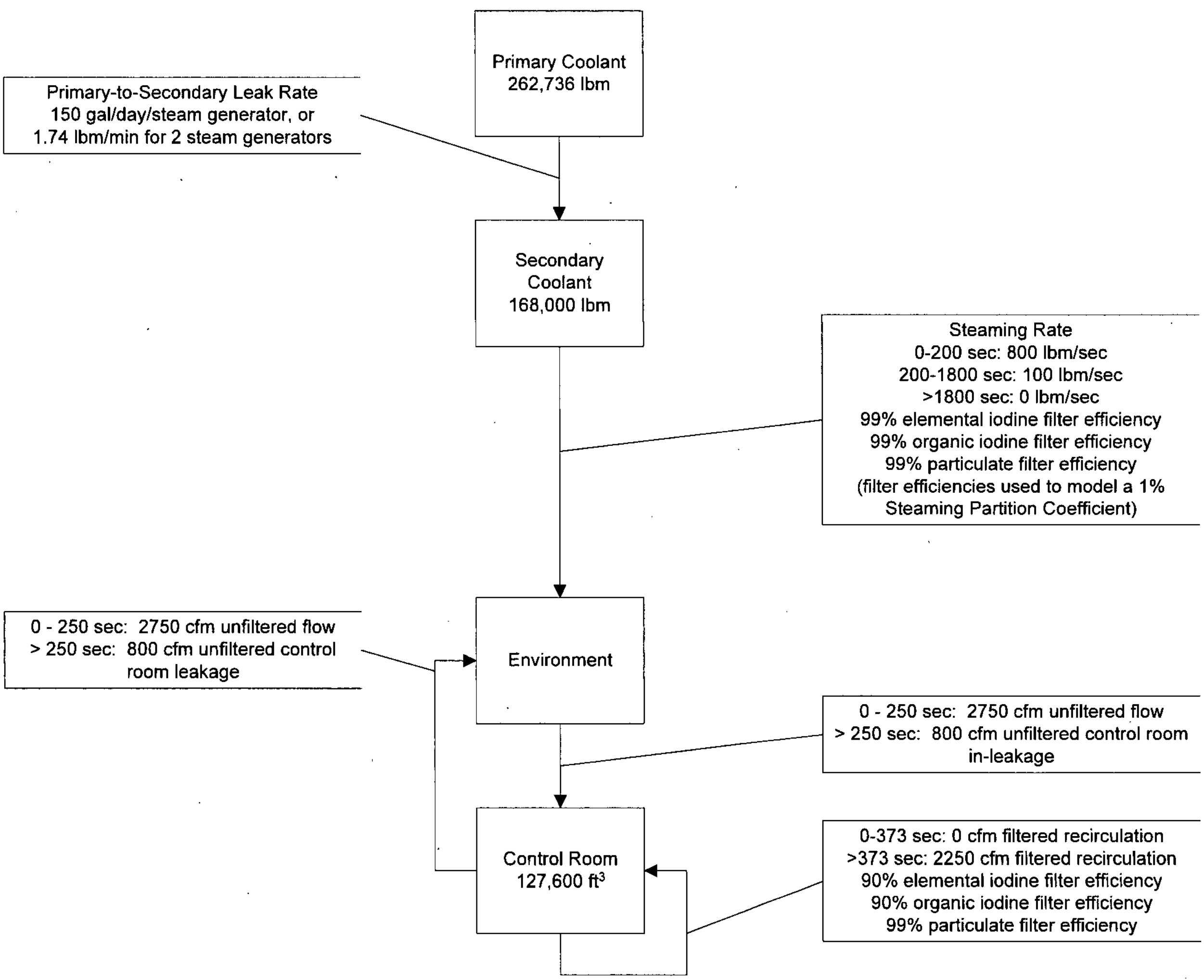
Figure 3.7-1 RADTRAD Model for Containment Airborne Releases



Secondary System Model

A secondary system release model assumes that 100% of the activity released from the fuel is completely dissolved in the primary coolant. This activity enters the secondary system via primary-to-secondary leakage and is then released to the environment. During the first 250 seconds of the accident the control room is not isolated.

Figure 3.7-2 RADTRAD Model for Secondary System Releases



3.7.5.2 Basic Data & Assumptions for REA

Changes have been made to the AST LRA. Table 3.7-2 provides a complete list of inputs and assumptions used to reanalyze the KPS LRA.

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
Core Power (MWt)	1782.6 (Licensed power of 1772 MWt with 0.6% uncertainty)	No Change	
Core Inventory (curies)	Licensed Uprated Core based on 1782.6 MWt multiplied by 1.06 (Table 3.2-3)	No Change	
Gap Fraction (%)		No Change	
Iodine	10		
Noble Gases	10		
Alkali Metals	12		
Initial Iodine Species in Containment (%)			
Elemental	4.85	No Change	
Methyl (organic)	0.15		
Particulate (aerosol)	95		
Rods in DNB (% of core)	15	No Change	
Melted Fuel (% of core)	0.375	No Change	
Power Peaking Factor	1.70	No Change	

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Activity Released into Containment (%)			
From Failed Fuel	100	No Change	
From Melted Fuel		No change	
Iodine	25		
Noble Gases	100		
Alkali Metals	100		
Activity Released into Primary Coolant (%)			
From Failed Fuel	100	No Change	
From Melted Fuel		No change	
Iodine	50		
Noble Gases	100		
Alkali Metals	100		
Initial Iodine Species in Containment (%)			
Elemental	4.85	No Change	
Methyl (organic)	0.15		
Particulate (aerosol)	95		
Primary-to-Secondary Leakage (gpd / SG)*	150	No Change	

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Iodine Species of Primary-to-Secondary Leakage (%) Elemental Methyl (organic) Particulate (aerosol)	 97 3 0	No Change	
Coolant Activity Concentration Prior to Accident Primary Iodine Primary Noble Gases Primary Alkali Metals Secondary Iodine Secondary Alkali	 60 µCi/gm DE I-131 Equiv. to 1% fuel defects Equiv. to 1% fuel defects 0.1 µCi/gm DE I-131 10% of Primary conc.	NONE	Per RG 1.183, appendix H, the source term for a PWR Rod Ejection accident only needs to consider the fuel damage postulated from the event. The CLB analysis shows inclusion of activity prior to the accident contributes less than 1% to the overall consequences.
Reactor Coolant Mass (gm)	1.22E+08	1.19E+08	Because no initial activity is assumed, lower RCS mass is conservative (higher concentration of failed fuel activity).

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
	Containment		
Containment Leak Rate (wt%/day) 0-24 hours >24 hours	 0.2 0.1	No Change	
Containment Leak Path Fractions <u>0-10 minutes</u> Through Shield Bldg Through Aux Bldg SV Direct to Environment	 0.0 0.10 0.90	No Change	
<u>10 minutes – 30 days</u> Through Shield Bldg Through Aux Bldg SV Direct to Environment	 0.89 0.10 0.01	No Change	
Shield Building Drawdown Time: (Tech Specs)	10 minutes	No Change	
Containment Volume (ft^3)	1.32E6	No Change	

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
	Containment Spray and Iodine Removal		
Containment Spray Removal	Not Credited	No Change	
Natural deposition (hr^{-1})	Not Credited	Power's Model set at the 10^{th} percentile	Per RG 1.183 Appendix H, natural deposition may be credited
	Shield Building		
Shield Building Annulus Volume (ft^3)	3.74E+05	No Change	
Shield Building Participation Fraction	0.5	No Change	
Shield Building Ventilation and Recirculation Iodine Filter Efficiency (%) Elemental Methyl (organic) Particulate (aerosol)	 90 90 99	 95 (includes safety factor of 2) 95 (includes safety factor of 2) 99	Conservative filter efficiencies for elemental and organic iodine were increased to be consistent with other accident analyses. Safety factor of 2 remains.
Shield Building Air Flow to Environment (cfm) 0-10 min 10-30 min >30 min	 0 6600 3100	No Change	

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Shield Building Recirculation Flow (cfm) 0-30 min >30 min	 0 2300	 No Change	
Auxiliary Building			
Participation with Auxiliary Building Volume or Hold-up	None	No Change	
Auxiliary Building Special Ventilation Iodine Filter Efficiency (%) Elemental Methyl (organic) Particulate (aerosol)	 90 90 99	 95 (includes safety factor of 2) 95 (includes safety factor of 2) 99	Conservative filter efficiencies for elemental and organic iodine were increased to be consistent with other accident analyses. Safety factor of 2 remains.
Secondary Release			
Primary to Secondary Leak rate (gpd from 2 SG)*	300	No Change	
Iodine Partitioning	PC = 100	No Change	
Alkali Metal Partitioning	PC = 100	No Change	

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Iodine chemical form of Primary-to-Secondary Leakage (%)	Elemental 97 Organic 3 Particulate 0	No Change	
Tube Uncovery.	No tube bundle uncovery assumed.	No Change	
Primary-to-Secondary Leak Duration (min)	30	No Change	
REA Parameters			
Safety Injection Signal (sec)	52.5	240	Delay of the SI signal is conservative. CLB assumption is based on a 2-inch diameter break. The REA is specified to have a smaller 1.6 inch diameter break. SI signal generated from a 1-inch diameter break is 240 seconds.
Steam Generator Liquid Mass (lbm/SG)	87,000	84,000	Minimum SG liquid volume used to minimize hold-up consistent with other secondary system release accidents.
Steam Release to Environment (lbm/sec) 0 – 200 sec 200 – 1800 sec >1800 sec	 800 100 0	No Change	

Table 3.7-2 Basic Data and Assumptions for REA					
Parameter or Assumption	**CLB Value**		**Proposed Value**		**Reason for Change**
Release point(s) Containment Pathway Secondary Release	 Containment / Shield Bldg Rx Building Stack Exhaust Aux Building Stack Exhaust "B" SG PORV		 No Change		
Release Termination (hr)	0.5		No Change		
EAB X/Q (sec/m^3) 0 – 2 hr	 2.232E-04		 1.76E-04		New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u>	<u>LPZ</u>	<u>Period</u>	<u>LPZ</u>	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
	0 – 2 hr	3.977E-05	0 – 8 hr	3.36E-05	
	2 – 24 hr	4.100E-06	8 – 24 hr	2.37E-05	
	1 – 2 day	2.427E-06	1 – 4 day	1.12E-05	
	2 – 30 day	4.473E-07	4 – 30 day	3.94E-06	
Control Room					
Control Room Volume (ft^3)	127,600		No Change		

Table 3.7-2 Basic Data and Assumptions for REA			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
Control Room Isolation (sec)	150	250	Control room isolation will occur 10 seconds following SI signal at 240 seconds.
Control Room Unfiltered Inleakage (cfm)	1000	800	To maintain consistency with all other radiological analyses. Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20)
Control Room Post Accident Recirculation system (CRPARS) Ventilation (min)	150	373	CRPARS initiation is assumed to be effective 133 seconds following SI. Based on 10 second delay to switchover from normal ventilation to emergency operation, 63 second delay in diesel loading of CRPARS, and 60 seconds to open recirculation dampers.

Table 3.7-2 Basic Data and Assumptions for REA						
Parameter or Assumption	**CLB Value**		**Proposed Value**		**Reason for Change**
Control Room HVAC Parameters (cfm)	<u>0 – 150 s</u>	<u>150 s – 30 d</u>	<u>0 – 250 s</u>	<u>250 s – 30 d</u>	Unfiltered inleakage is not assumed until the control room is isolated at 250 seconds at which time inleakage is assumed at 800 cfm, consistent with other DBA analyses.
Unfiltered Inleakage	0	1000	0	800	
Unfiltered Make-up Air	2750	0	2750	0	
Filtered Recirculation	0	2250	0	2250	
CRPARS Filter Efficiency (%)			No Change		
Elemental	90 (includes safety factor of 2)				
Organic	90 (includes safety factor of 2)				
Particulate	99				
Control Room X/Q (sec/m^3)	for all releases		<u>CR Intake</u>	<u>Inleakage</u>	New ARCON96 control room X/Q estimates (Table 1.3-4)
	0 - 8 hrs	2.93E-03	<u>0 – 2 hr</u>	<u>0 – 2 hr</u>	
Containment / Shield Bldg	8 – 24 hrs	1.73E-03	1.84E-03	1.74E-03	Prior to CR isolation (250 sec) the X/Q is to the CR intake. Post isolation, the inleakage X/Q represents the worst CR inleakage X/Q, by way of the turbine bldg, from each respective release point.
Rx Bldg Stack Exhaust	24 – 96 hrs	6.74E-04	4.88E-03	3.97E-03	
Aux Bldg Stack Exhaust	96 – 720 hrs	1.93E-04	3.67E-03	2.90E-03	
"B" SG PORV			3.96E-02	2.92E-02	For period values out to 720 hours, see Table 1.3-4

* *The density used to convert volumetric leak rates (gpd) to mass leak rates (lbm/hr) was consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications.*

3.7.6 REA Analysis Results

The total TEDE to the EAB, LPZ, and the Control Room from a RCCA Ejection Accident (REA) is summarized below in Table 3.7-3 for the containment and the secondary side release pathways. The containment pathway results in the highest dose consequences for both offsite and the control room. All doses are within the limits specified in Regulatory Guide 1.183 and 10 CFR 50.67.

Table 3.7-3 TEDE Results for the RCCA Ejection Accident

Location	TEDE (rem)	Limits (rem)
Containment Release Pathway		
EAB	0.2	6.3
LPZ	0.1	6.3
Control Room	0.8	5
Secondary Side Release Pathway		
EAB	0.1	6.3
LPZ	0.1	6.3
Control Room	0.5	5

3.8 Waste Gas Decay Tank Analysis

This section describes the methods employed and results of the Waste Gas Decay Tank failure (WGDT) design basis radiological analysis. The analysis assumes activity stored in a gas decay tank consists of the noble gases released from the processed coolant with only negligible quantities of the less volatile isotopes. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the KPS Control Room. The methodology used to evaluate the control room and offsite doses resulting from a WGDT accident included Standard Review Plan Branch Technical Position 11-5 (Reference 19), Regulatory Guide 1.24 (Reference 30), ARCON96-based control room atmospheric dispersion factors, PAVAND-based EAB and LPZ atmospheric dispersion factors, and Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

The current WGDT analysis credits the control room post accident recirculation system (CRPARS) in the determination of control room dose. New analyses have been performed to demonstrate that the CRPARS ventilation system is not required to maintain control room dose within acceptable limits.

3.8.1 WGDT Scenario Description

The Waste Gas Decay Tank (WGDT) accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in the waste gas storage system. Failure of a gas decay tank or associated piping could result in a release of this gaseous activity. The activity in a gas decay tank is taken to be the maximum amount that could accumulate from operation with cladding defects in 1 percent of the fuel elements. Per the guidance in Regulatory Guide 1.24, all gaseous radioactive material is assumed to release to the atmosphere over a 2 hour period.

3.8.2 WGDT Source Term Definition

The activity assumed in the WGDT analysis, previously calculated for the stretch power uprate and approved in Amendment No. 172, (Reference 11), represents the maximum activity of noble gases, xenon and krypton, accumulated over a full core cycle with 1% failed fuel. The maximum WGDT inventory for each nuclide is given in Table 3.8-1. These activities are extremely conservative compared to actual activity which would accumulate in the gas decay tanks based on revised reactor coolant activity limits. New Technical Specification requirements will limit reactor coolant activity (see Sections 2.3 and 3.4.2) to a small fraction of current requirements. The source term represented in Table 3.8-1 does not include the associated reductions that would be caused by operating at reduced RCS limits.

Table 3.8-1 Waste Gas Decay Tank Activity (Ci)

Nuclide	Activity in GDT Ci
Kr-85m	8.53E+01
Kr-85	2.39E+03
Kr-87	1.58E+01
Kr-88	1.08E+02
Xe-131m	5.20E+02
Xe-133m	4.76E+02
Xe-133	3.85E+04
Xe-135m	2.78E+01
Xe-135	6.68E+02
Xe-138	1.84E+00

3.8.3 WGDT Release Transport

The release scenario assumes the failure of a gas decay tank into the Auxiliary Building. No credit is taken for building volume dilution. The radioactive content of the tank is assumed to release over a two hour period. The release is modeled using the Auxiliary

Building Stack Exhaust as the release point to maximize the control room dose. The effluent resulting from the postulated event is assumed to release to the environment without continuous effluent radiation monitoring to automatically isolate and/or terminate the effluent release. No credit is taken for control room isolation, so the release is assumed to transport directly to the control room intake.

3.8.4 WGDT Atmospheric Dispersion Factors

3.8.4.1 WGDT Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The WGDT Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source point:

- Auxiliary Building Stack Exhaust

The control room X/Q determined for the Auxiliary Building Stack Exhaust to the Control Room Intake represents the highest value applicable to any source to receptor combination for the WGDT accident.

3.8.4.2 WGDT Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8).

3.8.5 WGDT Analysis Assumptions and Key Parameters

3.8.5.1 Method of Analysis

The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a WGDT at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

The total contents of the WGDT are assumed to be released over a 2 hour period. Assuming a removal rate of 3.45/hr will release essentially all (99.9%) of the gas decay tank contents over a two hour period. This equates to a release rate, for use in RADTRAD, of 8289%/day.

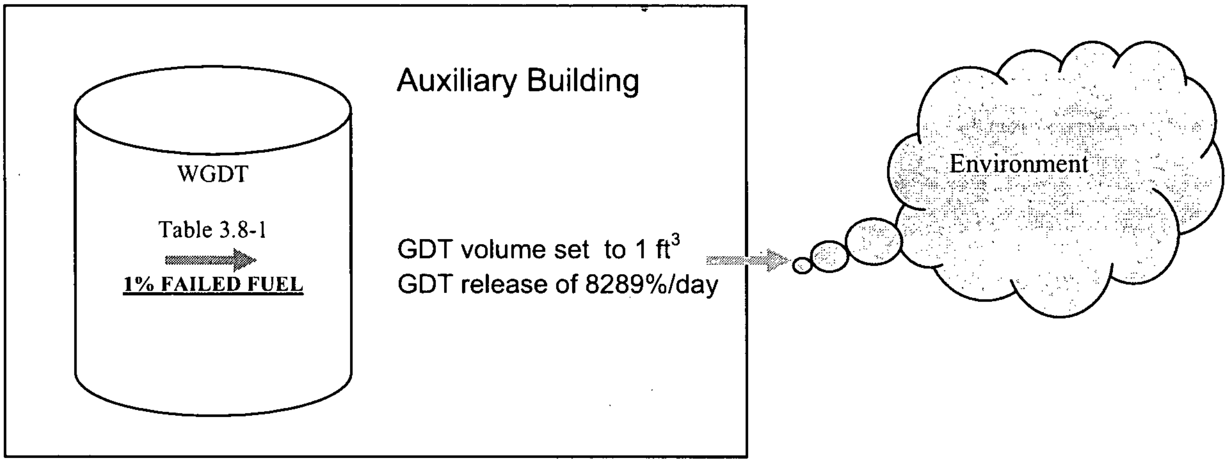
There are aspects of the WGDT analysis that require multiple RADTRAD models. Conservative combinations of control room ventilation rates (filtered recirculation and unfiltered inleakage) and control room isolation times (varied from quick isolation to delayed isolation to no isolation) were modeled in order to maximize control room dose and prove the CRPARS is not needed to maintain control room dose within acceptable limits (see Section 3.8.5.3).

A schematic shown in Figure 3.8-1 provides picture summary of the WGDT release to the environment modeled in RADTRAD.

3.8.5.2 Basic Data & Assumptions for WGDT

Changes have been made to the WGDT analysis. Table 3.8-2 provides a complete list of inputs and assumptions used to reanalyze the KPS WGDT event.

Figure 3.8-1 WGDT Radioactive Release Schematic



3.8.5.3 Assumptions to Maximize Control Room Dose

Control room dose is calculated with a combination of control room assumptions that maximize control room dose (i.e., 30 minute delayed isolation of the control room in conjunction with low unfiltered inleakage). This combination of assumptions maximizes control room occupant exposure during a short (2-hour) duration release and bounds the condition of no control room isolation. Intake of the radioactive material at the maximum control room ventilation intake rate will achieve a delayed equilibrium concentration as flow is both into and out of the control room. Delayed isolation of the control room and reducing the intake/outflow combination will trap the radioactivity within the control room and maximize the exposure to the occupants. Time sensitivity runs determined 30 minutes as the time that would maximize the control room dose. An unfiltered inleakage rate of 200 cfm was assumed to maximize control room dose. This rate is lower than one half of the minimum unfiltered inleakage air flow of 409 +/- 29 cfm measured by the American Society for Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004 (Reference 20). Unfiltered inleakage rates greater than 200 cfm produce lower control room dose due to the associated purge effect of the inflow.

Table 3.8-2 Basic Data and Assumptions for WGDT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Source Term			
WGDT Radiation Source (Curies)	Table 3.8-1	No Change	
Dose Consequence Multiplier	1.1	1.12	Source term adjustment factor to allow for fuel management variations. Previous multiplier allowed variation in cycle length of 493.6 ± 5% EFPD. The new higher multiplier accounts for a larger fuel management variation, similar to that required in the RSAC of 493.6 ± 10% EFPD.
RCS Coolant Activity (% failed fuel)	1	No Change	
Core Activity	Table 3.2-3	No Change	
WGDT Parameters			
Release Duration (min)	5	120	RG 1.24 allows the release period for the GDT rupture to be 2 hours.
Release Rate (%/day)	1.99E+05	8.289E+03	Tank contents are released over 2 hours rather than 5 minutes. Essentially all activity 99.9% is released over 2 hours at this reduced release rate.

Table 3.8-2 Basic Data and Assumptions for WGDT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
EAB X/Q (sec/m^3) 0 – 2 hr	2.232E-04	1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
Control Room			
Control Room Volume (ft^3)	127,600	No Change	
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
Control Room Post Accident Recirculation system (CRPARS) Ventilation (min)	0.5	NA	Control room post accident recirculation is not credited to maximize control room dose
Control Room Isolation (min)	0.5	30	The new proposed analysis assumes a bounding value of control room isolation time that will

Table 3.8-2 Basic Data and Assumptions for WGDT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
			maximize control room dose. Analyses performed assuming "NO" isolation produce control room consequences that are less than the proposed case with 30 minute isolation.
CRPARS Filter Efficiency (%) Elemental Organic Particulate	 90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	NA	Control room post accident recirculation is not credited to maximize control room dose.
Control Room Unfiltered Inleakage (cfm)	0	200	Low inleakage is assumed to maximize control room dose, but the grossly conservative assumption of no unfiltered inleakage was eliminated. Sensitivity cases show that higher inleakage will result in lower predicted dose. Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20) Minimum (ASTM) E741 tracer gas test = 409 ±29 cfm (Ref. 20)

Table 3.8-2 Basic Data and Assumptions for WGDT

Parameter or Assumption	CLB Value		Proposed Value		Reason for Change
Control Room HVAC Parameters (cfm)	**0 – 0.5 m**	**0.5 m – 30 d**	**0 – 30 m**	**30 m – 30 d**	Unfiltered inleakage is not assumed until the control room is isolated at 30 minutes at which time inleakage is assumed at 200 cfm (discussed above).
Unfiltered Inleakage	0	0	0	200	
Unfiltered Make-up Air	2750	0	2750	0	
Filtered Recirculation	0	2250	0	0	
Control Room X/Q (sec/m^3)	0 – 8 h	2.93E-3	0 – 2 h	3.67E-3	NEW ARCON96 X/Q values The highest calculated 0-2 hour X/Q value from the Auxiliary Building release pathway to the control room intake is from the Auxiliary Building Stack Exhaust.

3.8.6 WGDT Analysis Results

The results of the design basis WGDT analysis are presented in Table 3.8-3. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30-day duration of the event for the control room and the LPZ. The EAB and LPZ doses are calculated with RADTRAD and are compared with the applicable acceptance criteria specified in original licensing basis and Branch Technical Position 11-5, based on the earlier version of 10 CFR 20. Control Room dose is compared with the limits defined in General Design Criteria 19 (Reference 31) and applicable standards in RG 1.183.

Table 3.8-3 Dose Results for the WGDT Accident

Location	(rem)	Limits (rem)
EAB	0.1 (WB)	0.5 (WB)
LPZ	0.1 (WB)	0.5 (WB)
Control Room	0.4 (TEDE)	5 (TEDE)

The results in Table 3.8-3 represent the highest control room and offsite doses that would result from a WGDT accident using worst case scenario conditions. As discussed previously, the control room consequences above assume control room isolation and unfiltered inleakage assumptions that maximize control room dose. Control room dose in an unisolated control room will actually be less than the value listed in Table 3.8-3.

3.9 Volume Control Tank Rupture (VCT) Analysis

This section describes the methods employed and results of the Volume Control Tank rupture (VCT) design basis radiological analysis. The analysis assumes a failure of the VCT system that results in the release of the contents of the tank and additional releases from letdown flow until the letdown path is isolated. Doses were calculated at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ), and in the KPS Control Room. The methodology used to evaluate the control room and offsite doses resulting from a VCT accident included Standard Review Plan Branch Technical Position 11-5 (Reference 19), Regulatory Guide 1.24 (Reference 30), ARCON96-based control room atmospheric dispersion factors, PAVAND-based EAB and LPZ atmospheric dispersion factors, and Federal Guidance Reports (FGR) No. 11 and 12 dose conversion factors.

The current VCT analysis credits the control room post accident recirculation system (CRPARS) in the determination of control room dose. New analyses have been performed to demonstrate that the CRPARS is not required to maintain control room dose within acceptable limits.

3.9.1 VCT Scenario Description

The Volume Control Tank rupture (VCT) accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive noble gas and halogen activity contained in the VCT and additional releases from radioactivity contained in letdown flow until isolated. Rupture of the volume control tank is assumed to release all the contained noble gases and one percent of the halogen inventory of the tank plus that amount contained in the 88-gpm flow from the demineralizers, which would continue for up to five minutes before isolation.

3.9.2 VCT Source Term Definition

3.9.2.1 Activities

The activities assumed in the VCT analysis and shown in Tables 3.9-1, 3.9-2 and 3.9-3, were previously calculated for the stretch power uprate and approved in Amendment No. 172, (Reference 11). They represent the maximum activity of noble gases and halogens accumulated within the VCT and available for release. The inventory of gases in the tank is based on continuous operation with one percent fuel defects and without any purge of the gas space. The inventory of iodine in the tank is based on operation of the plant with one percent fuel defects and with 90 percent of the iodine removed by the letdown demineralizer. The maximum VCT inventory for each nuclide is given in Table 3.9-1. The concentration of noble gases in the letdown flow is listed in Table 3.9-2. In addition, a pre-accident iodine spike is assumed, although not required. The current assumption of a spike of 60 µCi/gm dose equivalent I-131 (DEI) is being maintained even though the limit is being reduced to 10 µCi/gm DEI in this license amendment request. The iodine concentration in the letdown flow is listed in Table 3.9-3.

The activities being assumed for the VCT rupture are extremely conservative compared to actual activity which would exist in the volume control tank and letdown line, based on revised reactor coolant activity limits. New Technical Specification requirements will limit reactor coolant activity (see Sections 2.3 and 3.4.2) to a small fraction of current requirements. The source terms represented in Tables 3.9-1, 3.9-2 and 3.9-3 do not include the associated reductions that would be caused by operating at reduced RCS limits.

3.9.2.2 Source Term Multiplier

The current licensing basis analysis for the VCT rupture includes a multiplier of 1.1 that is applied to the resulting calculated dose to allow for minor variations in fuel designs (e.g., core mass of 49.1 MTU +/- 10%, enrichment of 4.5 w/o +/- 10%, and cycle length of 493.6 EFPD +/- 5%). This 10% increase considers allowances for letdown flow variation (5.5%), VCT water volume variation (2.5%) and fuel management variation

(2%). Each of these allowances are individually conservative in their application. Together, they provide approximately a factor of two conservatism above the expected increase necessary to account for such variations.

To make the revised VCT analysis consistent with the assumptions applied to other analyses in this report that provide allowance for minor variations in fuel design, the variation in cycle length was increased from 493.6 EFPD +/- 5% to 493.6 EFPD +/- 10%. For noble gases and iodines, this variation has the effect of doubling the conservative fuel management variation from 2% to 4% based on sensitivity studies performed by Westinghouse. Therefore, the revised source term multiplier will increase from 1.1 to 1.12.

3.9.3 VCT Release Transport

The release scenario assumes the failure of the volume control tank or piping, releasing activity into the Auxiliary building. No credit is taken for building volume dilution. As a result of the accident, all of the noble gas in the tank and one percent of the iodine in the tank liquid is assumed to be released to the atmosphere over a period of 5 minutes. After event initiation, letdown flow to the volume control tank continues at the maximum flow rate of 88 gpm (maximum letdown flow plus 10-percent uncertainty) for 30 minutes when the letdown line is assumed to be isolated. The release is modeled using the Auxiliary Building Stack Exhaust as the release point to maximize the control room dose. The effluent resulting from the postulated event is assumed to release to the environment without continuous effluent radiation monitoring to automatically isolate and/or terminate the effluent release.

Table 3.9-1 Volume Control Tank Activity (Ci)

Nuclide	Activity in VCT (Ci)
Kr-85m	6.29E+01
Kr-85	7.35E+02
Kr-87	1.64E+01
Kr-88	8.85E+01
Xe-131m	2.07E+02
Xe-133m	2.21E+02
Xe-133	1.62E+04
Xe-135m	2.79E+01
Xe-135	4.52E+02
Xe-138	1.94E+00
I-131	8.69E-01
I-132	8.85E-01
I-133	1.30E+00
I-134	1.79E-01
I-135	7.09E-01

Table 3.9-2 Letdown Flow Noble Gas Concentration (µCi/gm)

Nuclide	Concentration (µCi/gm)
Kr-85m	1.73
Kr-85	8.60
Kr-87	1.13
Kr-88	3.28
Xe-131m	3.04
Xe-133m	3.44
Xe-133	242
Xe-135m	0.501
Xe-135	8.69
Xe-138	0.628

Table 3.9-3 Pre-Accident Iodine Spike Concentration based on 60 µCi/gm DEI

Nuclide	Concentration (µCi/gm)
I-131	46.8
I-132	47.6
I-133	69.8
I-134	9.7
I-135	38.2

3.9.4 VCT Atmospheric Dispersion Factors

3.9.4.1 VCT Control Room X/Qs

As described in Section 3.1, the onsite atmospheric dispersion factors were calculated using the ARCON96 code (Reference 5) and guidance from Regulatory Guide 1.194 (Reference 6). The VCT Control Room X/Qs listed in Table 1.3-4 were calculated for the following applicable KPS source point:

- Auxiliary Building Stack Exhaust

The control room X/Q determined for the Auxiliary Building Stack Exhaust to the Control Room Intake represents the highest value applicable to any source to receptor combination for the VCT accident.

3.9.4.2 VCT Offsite (EAB & LPZ) X/Qs

As described in Section 3.1, the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) atmospheric dispersion factors were revised and are listed in Table 1.3-3. These offsite atmospheric dispersion factors were generated using the PAVAND code (Reference 7) and guidance from Regulatory Guide 1.145 (Reference 8). The EAB and LPZ X/Q values were modeled using a ground-level release without credit for building wake to determine a conservative short-term diffusion estimate (X/Q).

3.9.5 VCT Analysis Assumptions and Key Parameters

3.9.5.1 Method of Analysis

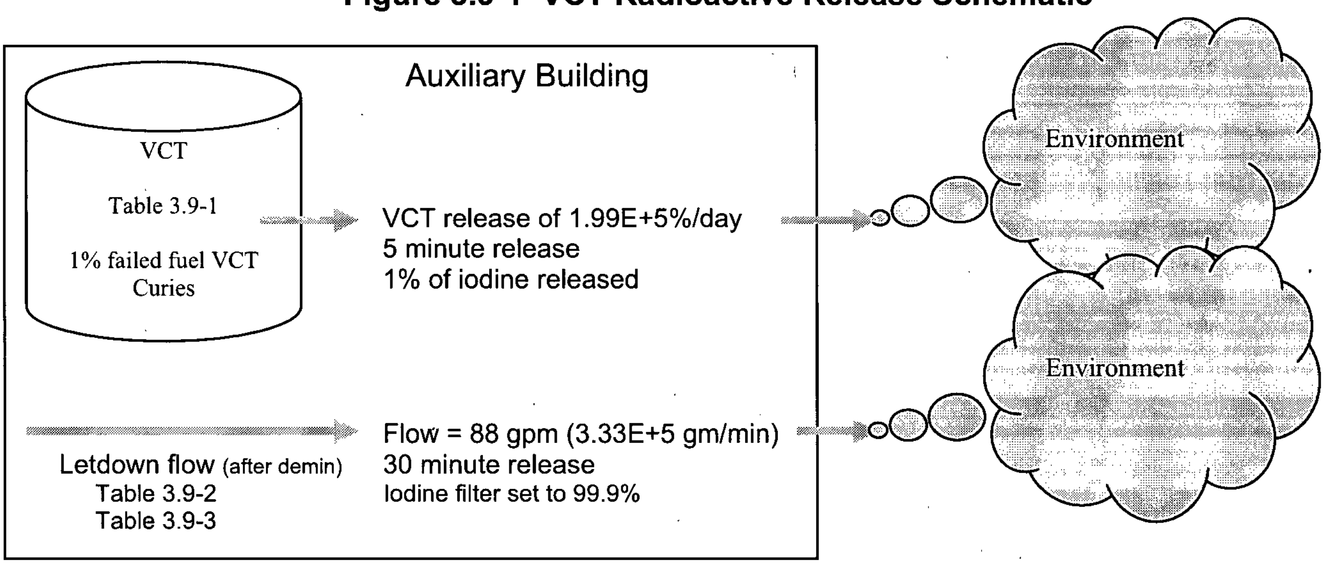
The RADTRAD-NAI code (Reference 3) is used to calculate the radiological consequences from airborne releases resulting from a VCT rupture at Kewaunee Power Station (KPS) to the EAB, LPZ, and Control Room.

The total contents of the VCT are assumed to release within 5 minutes. Assuming a removal rate of 82.9/hr will release essentially all (99.9%) of the tank contents over a five minute period. This equates to a release rate, for use in RADTRAD, of 1.99E+05%/day. The letdown release rate of 88 gallons per minute assumes 1% of the iodines after passing through the demineralizer in addition to a VCT tank DF of 10. This was modeled in RADTRAD as a filter with a filter efficiency set to 99.9% to mimic the iodine removal.

There are aspects of the VCT analysis that required multiple RADTRAD models. Conservative combinations of control room ventilation rates (filtered recirculation and unfiltered inleakage) and control room isolation times (varied from quick isolation to delayed isolation to no isolation) were modeled in order to maximize control room dose and prove the control room post accident recirculation system (CRPARS) is not needed to maintain control room dose within acceptable limits (see Section 3.9.5.3).

A schematic shown in Figure 3.9-1 provides an overall picture of the VCT release to the environment modeled in RADTRAD.

Figure 3.9-1 VCT Radioactive Release Schematic



3.9.5.2 Basic Data & Assumptions for VCT

Changes have been made to the VCT analysis. Table 3.9-4 provides a complete list of inputs and assumptions used to reanalyze the KPS VCT.

3.9.5.3 Assumptions to Maximize Control Room Dose

Control room dose was calculated with a combination of control room assumptions that maximize control room dose for the two release pathways modeled for a VCT rupture. One pathway is the tank rupture and near instantaneous release of tank radioactive contents within 5 minutes. The other pathway is the continual release of activity contained in letdown flow that will persist into the VCT and out of the ruptured tank until such time that letdown is isolated. Each pathway was evaluated for a condition of no control room isolation and delayed control room isolation. In both instances, delayed isolation produces higher control room dose. Results from both pathways were summed. The combination of assumptions for control room isolation and unfiltered inleakage rate that maximize control room dose were determined. The dose results, presented in Table 3.9-5, bound the condition of no control room isolation.

For the VCT rupture pathway, the set of control room assumptions that produced the highest control room dose was a 2.5 minute delayed isolation of the control room in conjunction with low unfiltered inleakage. For the letdown line release pathway that persists for 30 minutes until letdown is isolated, the set of control room assumptions that will maximize control dose was a 30 minute isolation of the control room in conjunction with low unfiltered inleakage. This combination of assumptions maximize the dose to inhabitants of the control room. Intake of the radioactive material at the maximum control room ventilation intake rate with delayed isolation of the control room and reduced intake/outflow will trap the radioactivity within the control room and maximize the exposure to the occupants. Time sensitivity runs determined that 2.5-minute isolation maximized the 5-minute VCT rupture pathway scenario and 30-minute isolation maximized the 30-minute letdown pathway scenario. Assuming no control room isolation or isolation prior to or after the times listed, produce lower dose consequences. Unfiltered inleakage rate sensitivity runs showed that control room dose is maximized by assuming a low unfiltered inleakage rate. An unfiltered inleakage rate of 200 cfm was determined to maximize control room dose. This rate is lower than one half of the unfiltered inleakage air flow of 409 +/- 29 cfm measured by the American Society for

Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004 (Reference 20). Unfiltered inleakage rates greater than 200 cfm produce lower control room dose due to the associated purge effect of the inflow.

The design assumptions for the VCT rupture analysis that maximize dose and demonstrate that the control room post accident recirculation system is not needed are listed in Table 3.9-4.

Table 3.9-4 Basic Data and Assumptions for VCT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
	Source Term		
VCT Radiation Source (Curies)	Table 3.9-1	No Change	
Letdown Line RCS Noble Gas Concentration (μCi/gm)	Table 3.9-2	No Change	
Letdown Line Pre-Accident Iodine Spike Concentration (μCi/gm)	Table 3.9-3 [conservatively based on spike of 60 μCi/gm DEI]	No Change	
Iodine Release from VCT and Letdown Line (%)	1	No Change	
Source Term Multiplier	1.1	1.12	Source term adjustment factor to allow for fuel management variations. Previous multiplier provided by Westinghouse allowed variation in cycle length of 493.6 ± 5% EFPD. The new higher multiplier accounts for a larger fuel management variation, similar to that required in the RSAC of 493.6 EFPD ± 10% (see Section 3.9.2.2).
RCS Coolant Activity (% failed fuel)	1	No Change	
Core Activity	Table 3.2-3	No Change	

Table 3.9-4 Basic Data and Assumptions for VCT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Demineralizer Iodine DF for Letdown Flow prior to VCT	10	No Change	
VCT Parameters			
Release Duration (min) VCT Letdown	 5 5	 No Change 30	
Release Rate (%/day)	1.99E+05	No Change	
EAB X/Q (sec/m^3) 0 – 2 hr	 2.232E-04	 1.76E-04	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
LPZ X/Q (sec/m^3)	<u>Period</u> <u>LPZ</u> 0 – 2 hr 3.977E-05 2 – 24 hr 4.100E-06 1 – 2 day 2.427E-06 2 – 30 day 4.473E-07	<u>Period</u> <u>LPZ</u> 0 – 8 hr 3.36E-05 8 – 24 hr 2.37E-05 1 – 4 day 1.12E-05 4 – 30 day 3.94E-06	New PAVAND X/Q values (see Table 1.3-3 and Section 3.1.2)
Letdown Flow (gpm)	88	No Change	
RCS Mass (grams)	1.192E+08	No Change	
Control Room			
Control Room Volume (ft^3)	127,600	No Change	

Table 3.9-4 Basic Data and Assumptions for VCT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
Normal Ventilation Unfiltered Makeup Air Flow (scfm)	2,750	No Change	
Filtered Recirculation Air Flow (scfm)	2,250	No Change	
Control Room Post Accident Recirculation system (CRPARS) Ventilation (min)	0.5	NA	Control room post accident recirculation is not credited to maximize control room dose.
Control Room Isolation (min) VCT Letdown	 0.5 0.5	 2.5 30	The new Proposed analysis assumes a bounding value of control room isolation times that will maximize each pathway control room dose. Analyses performed assuming "NO" isolation produce control room consequences that are lower.
CRPARS Filter Efficiency (%) Elemental Organic Particulate	 90 (includes safety factor of 2) 90 (includes safety factor of 2) 99	NA	Control room post accident recirculation is not credited to maximize control room dose.
Control Room X/Q (sec/m^3)	0 – 8 h 2.93E-03	0 – 2 h 3.67E-03	NEW ARCON96 X/Q values The highest calculated 0-2 hour X/Q value of from any possible release pathway from the Auxiliary Building to the control room intake is from the Auxiliary Building Stack

Table 3.9-4 Basic Data and Assumptions for VCT			
Parameter or Assumption	**CLB Value**	**Proposed Value**	**Reason for Change**
			Exhaust.
EAB and LPZ X/Q (sec/m^3)	EAB 2.232E-04 LPZ (0-2 hr) 3.977E-05 (2-24 hr) 4.100E-06 (1-2 day) 2.427E-06 >2 day 4.473E-07	Table 1.3-3	NEW PAVAND X/Q values
Control Room Unfiltered Inleakage (cfm)	0	200	The grossly conservative assumption of no unfiltered inleakage was raised but still remains lower than measured inleakage. Low inleakage is assumed to maximize control room dose. Sensitivity cases show that higher inleakage will result in lower predicted dose. Maximum (ASTM) E741 tracer gas test = 447±51 cfm (Ref. 20)

Table 3.9-4 Basic Data and Assumptions for VCT					
Parameter or Assumption	**CLB Value**		**Proposed Value**		**Reason for Change**
Control Room HVAC Parameters (cfm)					Unfiltered inleakage of 200 cfm is not assumed until the control room is isolated.
VCT	**0 – 0.5 m**	**0.5 m – 30 d**	**0 – 2.5 m**	**2.5 m – 30 d**	
Unfiltered Inleakage	0	0	0	200	
Unfiltered Make-up Air	2750	0	2750	0	A conservative combination of control room isolation times and inleakage assumptions were used to maximize control room dose.
Filtered Recirculation	0	2250	0	0	
Letdown	**0 – 0.5 m**	**0.5 m – 30 d**	**0 – 0.5 hr**	**0.5 hr – 30 d**	
Unfiltered Inleakage	0	0	0	200	Analyses performed assuming "NO" isolation produce control room consequences that are lower.
Unfiltered Make-up Air	2750	0	2750	0	
Filtered Recirculation	0	2250	0	0	

3.9.6 VCT Analysis Results

The results of the design basis VCT analysis are presented in Table 3.9-5. These results report the calculated dose for the worst 2-hour interval (EAB), and for the assumed 30-day duration of the event for the control room and the LPZ. The EAB and LPZ doses are calculated with RADTRAD and are compared with the applicable acceptance criteria specified in original licensing basis and Branch Technical Position 11-5, based on the earlier version of 10 CFR 20. Control Room dose is compared with the limit specified in General Design Criteria 19 (Reference 31) and applicable standards in RG 1.183.

Table 3.9-5 Dose Results for the VCT Accident

Location	(rem)	Limits (rem)
EAB	0.1 (WB)	0.5 (WB)
LPZ	0.1 (WB)	0.5 (WB)
Control Room	0.6 (TEDE)	5 (TEDE)

The results in Table 3.9-5 represent the highest control room and offsite doses that would result from a VCT accident using worst case scenario conditions. As discussed previously, the control room consequences above were calculated using control room isolation and unfiltered inleakage assumption combinations that will maximize control room dose. Control room dose in an unisolated control room would be less than the value listed in Table 3.9-5.

4.0 ADDITIONAL DESIGN BASIS CONSIDERATIONS

In addition to the explicit evaluation of radiological consequences that had direct impact from the changes associated with this request, other areas of plant design were also considered for potential impacts. The evaluation of these additional design areas is documented below.

4.1 Risk Impact of Proposed Changes

The proposed changes associated with implementation of the revised design basis radiological analyses for Kewaunee Power Station have been considered for their risk effects. A discussion of these considerations is presented below.

The proposed changes are presented here for convenience; these changes are described in report Section 2:

a. Revised Meteorological X/Q Values for Off-site and Control Room Receptors
b. Use of the RADTRAD-NAI Code to analyze Dose Consequences
c. Reduction in Maximum RCS Coolant Activity Limits
d. Reduction in SG Secondary Coolant Activity Limit
e. Isolation of the Control Room prior to moving Recently Irradiated Fuel
f. Refueling Operation Requirements to allow Open Containment Penetrations
g. Elimination of R-23 Credit for Control Room Isolation
h. Revise Technical Specification Definition of Dose Equivalent I-131
i. Changes in Design and License Basis Assumptions

Item a — The change in X/Q values has a direct effect on calculated dose consequences. The new values were calculated pursuant to the guidance of Regulatory Guides 1.145 and 1.194, respectively. Their use in design basis analyses assure that the resulting consequences contain sufficient conservatism due to atmospheric dispersion such that the value is not exceeded by more than 5.0 percent of the time. This change has no impact upon plant risk.

Item b RADTRAD-NAI designed after the ITSC version of RADTRAD developed for the NRC, has been previously found to be acceptable for use in dose calculations. Its use has no impact upon plant risk.

Item c The reduction in maximum allowed RCS coolant activity in Technical Specifications will cause a commensurate reduction in potential dose consequences as a result of RCS releases. Reducing RCS concentration has no impact upon plant risk.

Item d The reduction in maximum allowed secondary side activity will cause a commensurate reduction in potential dose consequences as a result of secondary side releases. Reducing secondary side concentration has no impact upon plant risk.

Item e Control room isolation will be required prior to moving recently irradiated fuel. This measure was necessary to eliminate credit for R-23 and maintain control room dose within limits. Having the control room isolated does not impact plant risk.

Item f Allowing containment penetrations to be open during movement of recently irradiated fuel has been shown to result in acceptable off-site consequences. The design analysis assumes the containment remains open for the entire 2-hour duration of the fuel handling event. Being under Technical Specification required Administrative Control, the ability and likelihood for closure of open containment penetrations, in the event of an accident, is increased. Closure of penetrations is an additional defense-in-depth, not credited – but available if conditions warrant. Allowing penetrations to be open during refueling provides flexibility in outage scheduling and additional comfort to workers. Open penetrations have no impact upon plant risk.

Item g The elimination of control room isolation credit from the control room inlet monitor R-23 removes reliance for a safety function performed by instrumentation that is not redundant, not safety grade, and provides incomplete isolation of the control room. Credit for R-23 currently exist for the FHA and LRA. With the requirement to require control room isolation prior to movement of recently irradiated fuel, plant risk is not impacted and the FHA control room consequences are acceptable. Crediting Operator action to isolate the control room within 1-hour after LR event will result in acceptable dose consequences. Current license basis credits Operator action within 45 minutes of a LRA if R-23 fails to perform its safety function. Credit for Operator action has been extended to 1-hour, reducing timing burden on the Operator. Extending the allowed time to isolate control room following a LRA does not impact plant risk. No longer crediting R-23 for control room isolation is acceptable and removes future vulnerability by reliance on non-safety grade instrumentation to perform a safety function.

Item h Changing the Technical Specification definition of DEI allows the use of dose conversion factors from FGR 11. These dose conversion factors have been previously found to be acceptable for use in dose calculations. This change has no impact upon plant risk from severe accident scenarios.

Item i The changes in design and license basis assumptions have been evaluated for the full spectrum of USAR Chapter 14 design basis analyses. The changes proposed in congregate form have been demonstrated to result in acceptable off-site and control room dose consequences. All assumptions have been validated and are presented for approval with associated discussions on methods and inputs. The changes are deemed safe and do not pose an impact on plant risk.

The revised assessments of the radiological consequences due to design basis accidents listed in the KPS USAR, using the AST methodology and proposed assumptions and inputs, conclude that the EAB, LPZ, and Control Room doses are within the limits of 10 CFR 50.67 and within the limits of Regulatory Guide 1.183. The results of this proposed amendment demonstrate that there will be no adverse impact on public health and safety.

4.2 Impact Upon the Emergency Plan

This proposed revision to Technical Specifications and USAR design basis analyses will replace the existing radiological licensing basis upon approval. The current Emergency Action Levels (EAL) for KPS implement the NEI 99-01 Rev. 4 (Reference 29) guidance. EAL limits (SU 4.1 and SU 4.2 for hot conditions and CU 5.1 for cold shutdown) apply criteria that relate reactor coolant sample activity and associated radiation monitor readings to provide indication of fuel clad integrity. These limits, if exceeded, are considered to be a potential degradation in the level of safety of the plant and a potential precursor of more serious problems. The current limits are tied to Technical Specification 3.4.16 limits on RCS activity and spikes. With the proposed changes to TS 3.4.16 to reduce RCS activity limits, corresponding changes to these three EAL limits will be necessary to maintain the same level of effectiveness and maintain the same technical basis. SU 4.1 and CU 5.1 letdown radiation monitor (R-9) limits based on an RCS concentration of 1.0 μCi/gm DE I-131 will need to be reduced by a factor of ten to correspond to the proposed RCS activity limit reduction. This new limit remains sufficiently above normal background readings on R-9 to provide indication of a degraded fuel condition. Likewise, SU 4.2 RCS activity limits which are based on TS 3.4.16 will need to be revised to correspond with the new proposed technical specification limits for RCS activity.

Other than RCS activity limit reductions, design basis source terms were unaffected by this license amendment request. In addition, revised design basis X/Q dispersion factors are not used by the emergency plan. Therefore, beyond the above identified change to the EALs, existing emergency plan procedures and dose assessment tools and models are unaffected by the changes proposed in this request.

5.0 Conclusions

The proposed changes in Technical Specifications, design assumptions, and offsite and control room X/Qs have been incorporated into the reanalysis of radiological effects from eight key accidents for KPS. The analysis results from the reanalyzed events meet all of the acceptance criteria as specified in 10 CFR 50.67, RG 1.183, and BTP 11-5.

6.0 References

1. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," USNRC, Office of Nuclear Regulatory Research," July 2000.

2. 10 CFR 50.67, "Accident Source Term"

3. Software - RADTRAD-NAI Version 1.1a(QA), Numerical Applications Inc.

4. NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," USNRC, June 1997, S.L. Humphreys et al.

5. NUREG/CR-6331, Rev. 1, "Atmospheric Relative Concentrations in Building Wakes, ARCON96," USNRC, 1997.

6. Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, June 2003

7. Software – PAVAND Version 1-00, Atmospheric Dispersion Model.

8. Regulatory Guide 1.145, Revision 01, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," February 1983.

9. Letter K-88-032, K.E. Perkins, USNRC to D.C. Hintz, WPSC; "Application of Leak-Before-Break Technology as a Basis for Kewaunee Nuclear Power Plant Steam Generator Snubber Reduction," 2/16/88.

10. Letter from John Lamb (NRC) to Tom Coutu (NMC) transmitting Issuance of Alternate Source Term Amendment # 166 (TAC No. MB4596) and Safety Evaluation Report dated March 17, 2003.

11. Letter from John Lamb (NRC) to Tom Coutu (NMC) transmitting Issuance of Stretch Power Uprate Amendment #172 (TAC No. MB9031) and Safety Evaluation Report dated February 27, 2004.

12. License Amendment Request 211, "Radiological Accident Analysis and Associated Technical Specifications Change," dated January 30, 2006, (ML060540217).

13. RAI Response Regarding License Amendment Request – 211 "Radiological Accident Analysis and Associated Technical Specifications Change," dated January 23, 2007 (ML070240543).

14. Letter from R. F. Kuntz (NRC) to D. A. Christian (Dominion) transmitting issuance of Radiological Accident Analysis Amendment #190 (TAC No. MC9715) and Safety Evaluation Report dated March 8, 2007.

15. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA 520/1-88-020, Environment Protection Agency, 1988.

16. Federal Guidance Report No. 12, "External Exposures to Radionuclides in Air, Water and Soil," EPA 420-r-93-081, Environmental Protection Agency, 1993.

17. Not Used.

18. Regulatory Guide 1.23, Revision 1, "Meteorological Monitoring Programs for Nuclear Power Plants," March 2007.

19. NRC Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases due to a Waste Gas System Leak or Failure," Rev 0, July 1981.

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

21. NUREG-0800, "Standard Review Plan," Section 6.4, "Control Room Habitability System," Revision 2, July 1981.

22. NUREG-0800, "Standard Review Plan," Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," U.S. Nuclear Regulatory Commission, Revision 2, December 1988.

23. NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays," June 1993.

24. KPS Drawing M-358, Revison L, "Reactor Building Piping – Internal Containment Spray," August 19, 2008.

25. NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," July 1996.

26. BNP-100, "Iodine Removal from Containment Atmospheres by Boric Acid Spray," July 1970.

27. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident," December 1997.

28. NSAL-00-004, Westinghouse Nuclear Safety Advisory Letter dated March 7, 2000, "Non-conservatisms in Iodine Spiking Calculations."

29. NEI 99-01, Revision 4, "Methodology for Development of Emergency Action Levels," January 2003.

30. Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)," March 1972.

31. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19 – Control Room (GDC 19).

32. Procedure GNP-05.16.06, "Validation of Time Dependent Operator Actions," Revision 6.

33. RAI Response Regarding Millstone Unit 3 Stretch Power Uprate License Amendment Request Response to Question AADB-07-0107, dated January 18, 2008 (ML080280375).

34. License Submittal, North Anna Power Station Units 1 and 2, "Proposed Technical Specification Change and Supporting Safety Analyses Revisions to Address Generic Safety Issue 191," dated October 3, 2006, (ML062850195).

ATTACHMENT 5

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

EVALUATION OF NEW PROPOSED MANUAL ACTIONS

KEWAUNEE POWER STATION
DOMINION ENERGY KEWAUNEE, INC.

EVALUATION OF PROPOSED NEW MANUAL ACTIONS

Introduction

In accordance with the revised RAA provided in Attachment 4, DEK is proposing two manual actions to ensure post-accident dose is maintained within limits. The revised RAA credits these manual actions to limit consequences of the Fuel Handling Accident (FHA) and Locked Rotor Accident (LRA). The proposed manual actions are as follows:

1. The revised RAA credits manual operator action to isolate the control room envelope (CRE) within one hour after initiation of an LRA. This manual action is required to compensate for the proposed TS changes that would discontinue credit for CRE 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 FAA.

1.0 Manual Action for the Locked Rotor Accident

This proposed manual action would require operator action to isolate the CRE within one hour after initiation of an LRA. This manual action is required to compensate for the proposed TS changes that would discontinue credit for CRE auto-isolation using a high radiation signal from R-23. R-23 is a single channel non-safety related instrument, and therefore DEK has proposed not crediting this radiation monitor in the revised radiological analyses. The LRA scenario is described in Attachment 4, section 3.6.

Verification of successful action is provided in KPS Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection." EOP E-0 provides direction regarding which status lights and annunciators will be illuminated if SI is actuated. In addition, verification of CRE isolation and CRPAR initiation is provided by observing status lights on the control board for the CRPAR fan and Control Room Air Conditioning (CRAC) fan. If an automatic actuation of SI does not occur during this accident, the operators are directed to manually initiate both trains of SI once subcooling is lost. This action will isolate the CRE and start both CRPAR trains.

If the proposed manual action is accomplished within one hour of an LRA occurring, then control room doses will be maintained within the limits specified in 10 CFR 50.67.

2.0 Manual Actions for the Fuel Handling Accident

This proposed manual action is based on the premise that upon initiation of a postulated FHA, the CRE will have been previously manually isolated in accordance with a proposed new Note in TS 3.7.10, "Control Room Post Accident Recirculation (CRPAR) System." The proposed new manual action consists of initiating one train of the CRPAR system within 20 minutes after the occurrence of a FHA. The revised FHA analysis is provided in Attachment 4, Section 3.3. The revised FHA analysis assumes that the CRE is isolated prior to moving recently irradiated fuel as required by TS 3.7.10. Upon occurrence of a FHA, the analysis assumes manual operator action to initiate one train of the CRPAR system. This manual operator action must be completed within 20 minutes following a FHA to ensure control room occupant dose remains within the limits specified in 10 CFR 50.67.

Control room operators would be promptly notified of a FHA by either of the following methods. These methods provide multiple and diverse means of alerting control room operators to the occurrence of a FHA.

1. KPS Procedure NF-KW-RRF-014, "Fuel Movement During a Refueling Outage," requires direct communication be maintained between the control room and the containment operating floor whenever changes in core geometry are taking place. This ensures that control room operators would be promptly alerted if an FHA event occurs.
2. The KPS Technical Requirements Manual (TRM) 8.9.4, "Radiation Monitoring During Refueling Operations," requires continuous monitoring of radiation levels in the containment and spent fuel pool areas during refueling operations. TRM 8.9.4 is met by requiring radiation monitors R-2, R-5, R-12 and R-21 to be operating during refueling operations. Each of these radiation monitors alarms in the control room.

The proposed changes to TS 3.7.10 (see Attachment 1, Section 2.2.2) would require the CRE be isolated with no fresh air being supplied to the control room (outside air dampers closed and CRPAR fan off) during movement of recently irradiated fuel assemblies. In this configuration, the proposed manual action consists of initiating one train of the CRPAR system. One train of the CRPAR system is initiated by turning either the A (ES-46545) or B (ES-46546) control room hand switch for CRPAR Recirculation Fan to the ON position. Then, using control switch ES 40030 recirculation damper ACC3A is opened, or using control switch ES40031 recirculation damper ACC3B is opened, depending on which train is being started. Proper operation of the train would be verified by the operator using status lights on the control board for the CRPAR fan and CRAC fan. The revised RAA assumes one train of the filtration/recirculation system is placed in operation within 20 minutes of FHA initiation.

The proposed changes to TS 3.7.10 would require that the CRE be isolated with no fresh air being supplied to the control room (outside air dampers closed and CRPAR fan off) during movement of recently irradiated fuel assemblies. However, this TS can be modified with application of the existing TS 3.7.10 Note which permits the CRE boundary to be opened intermittently under administrative controls. In this less likely alignment, the manual action would consist of closing one outside air damper, in addition to initiating one train of the CRPAR system. In this configuration, the revised RAA assumes one train of the filtration/recirculation system is placed in operation within 20 minutes of FHA occurrence and the CRE is fully isolated.

In this configuration, the operator must release the control room switches for outside air dampers ACC-2 (ES-46827) and ACC-1A/1B (ES-46833) to perform the required alignment. To align fresh air to the control room with the CRPAR system operating (either train) requires the operator to hold the selector switch for ACC-1A/1B to the "Normal" (ACC-1A) or "Alt" (ACC-1B) position and hold the control switch for ACC-2 in the "Open" position. Since the operator is required to hold the switches in position, this configuration would be used sparingly and for short durations to provide fresh air to the control room. Therefore, this is considered an infrequent control room ventilation system configuration. Each of these control switches are spring return to "Auto" position and are interlocked to close the dampers when either CRPAR fan is running. Therefore, when the operator releases the control switches, the CRE will return to the isolated configuration with at least one CRPAR fan running. No further actions would be required except to verify the correct alignment.

3.0 Acceptability of Proposed Manual Actions

The NRC has provided guidance regarding the requirements for use of manual actions. NRC RIS 2005-20, Revision 1, Section C.5 (reference 1), discusses the conditions under which temporary manual actions may be used in lieu of automatic actions in support of operability. NRC Information Notice (IN) 97-78 (reference 2) alerted licensees to the importance of considering the effects on human performance of such changes made to plant safety systems. Information Notice 97-78 states:

> *"The original design of nuclear power plant safety systems and their ability to respond to design-basis accidents are described in licensees' FSARs [final safety analysis reports] and were reviewed and approved by the NRC. Most safety systems were designed to rely on automatic system actuation to ensure that the safety systems were capable of carrying out their intended functions. In a few cases, limited operator actions, when appropriately justified, were approved. Proposed changes that substitute manual action for automatic system actuation or that modify existing operator actions, including operator response times, previously reviewed and approved during the original licensing review of the plant will, in all likelihood, raise the possibility of an unreviewed safety question (USQ). Such changes must be evaluated under the criteria of 10 CFR 50.59 to determine whether a USQ is involved and whether NRC review and approval is required before implementation.... In the NRC staff's experience, many of the changes* [involving operator actions] *proposed by licensees do involve a USQ."*

It is recognized that the NRC updated 10 CFR 50.59, to remove the USQ wording. Nonetheless, the intent of IN 97-78 is still pertinent. That is, licensees still need to submit many of the changes in operator actions to the NRC for review and approval in accordance with 10 CFR 50.59.

The guidance presented in NUREG-1764 (reference 3) can be used to address safety-related operator actions (SROAs), as well as other required operator actions. The American National Standards Institute/American Nuclear Society defines "safety-related operator action" in ANSI/ANS-58.8-1994, as follows:

> *"A manual action required by plant emergency procedures that is necessary to cause a safety-related system to perform its safety-related function during the course of any DBE (design-basis event). The successful performance of a safety-related operator action might require that discrete manipulations be performed in a specific order."*

Per NUREG-1764 changes in human actions (HAs) (synonymous with the term "operator actions") result from the following types of plant activities:

- Plant modifications.
- Procedure changes.
- Equipment failures.
- Justifications for continued operations (JCOs)[1].
- Identified discrepancies in equipment performance or safety analyses.

NUREG-1764 provides guidance for the review of human actions. This document provides guidance for use in determining the appropriate level of human factors engineering (HFE) review of HAs based upon their risk-importance. This guidance uses a graded, risk-informed (RI) approach consistent with RG 1.174, Rev. 1 (reference 4).

This guidance uses a two-phased approach to reviewing HAs. Phase 1 is a risk screening and analysis of the affected HAs identified to determine their risk-importance and the level of HFE review that is appropriate in Phase 2. Phase 2 is an HFE review of those HAs that are found to be risk-important.

3.1. Phase I – Risk Screening

KPS has elected to provide this application using non-risk informed (non-RI) analysis techniques. The non-RI screening process consists of the following steps:

1. Verify that the non-RI change request is appropriate.
2. Assess safety-significance of the HAs.
3. Qualitatively assess the safety-significance of HAs involved in the change request.

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[1] NOTE: The term JCO is no longer recognized by the NRC as valid. Per RIS 2005-20 (reference 29), "An SSC that is determined to be operable but degraded or nonconforming is considered to be in compliance with its TS LCO, and the operability determination is the basis for continued operation."

4. Make an integrated assessment of HA safety-significance to determine the appropriate level of HFE review (i.e., Level 1, 2, or 3).

The assessment of these four steps is provided below. Requirements are in normal text and responses are provided in italics text.

1. The non-RI change request is appropriate

NUREG-0800 (SRP) Chapter 19.2, (reference 5), Appendix D addresses the use of risk information in reviewing requests containing manual actions in non-RI license amendments. In accordance with the guidance, the risk implications of a non-RI submittal would warrant further risk informed analysis if the submittal:

- Significantly changes the allowed outage time (e.g., outside the range previously approved at similar plants), the probability of the initiating event, the probability of successful mitigative action, the functional recovery time, or the operator action requirement;

  Response: *There is no significant change to allowed outage time. The proposed manual actions are limited to the response to the FHA and LRA. There is no change to the allowed outage time of any equipment designed to mitigate these accidents. DEK is proposing new TS requirements to isolate the control room prior to movement of recently irradiated fuel assemblies. This new requirement simplifies the necessary manual action in the event of a fuel handling accident. Furthermore, the manual actions proposed do not change the probability of any initiating event or the probability of successful mitigation of events as discussed in Attachment 4. Finally, the proposed operator actions do not change the functional recovery time of any other accident scenario or change other operator actions required to recover from another accident.*

- Significantly changes functional requirements or redundancy;

  Response: *The proposed manual actions do not significantly change functional requirements or redundancy. Operation of the CRPAR system is required for radiological accidents and a CRE isolation is assumed. The proposed manual actions do not change the system operation. CRPAR system starting, filtering and redundancy requirements are not changed. CRE isolation redundancy requirements have not changed. There is no change in redundancy as both CRPAR trains are still required to be operable as well as the CRE per the proposed TS changes. Therefore, there is no change in the functional requirements or redundancy for the CRPAR system and CRE isolation.*

- Significantly changes operations that affect the likelihood of undiscovered failures;

Response: *The proposed operator action does not significantly change operations that affect the likelihood of undiscovered failures. Failures of the CRPAR system or dampers to isolate are indicated by lights in the control room. Therefore, operators will be made aware of a failure of the CRPAR system or CRE through these lights. The manual actions proposed herein would not mask or hide any undiscovered failures of the CRPAR system or the CRE.*

- Significantly affects the basis for successful safety function;

  Response: *The KPS current licensing bases (CLB) relied on a non-redundant, non-safety related radiation monitor to initiate the CRPAR system and perform a partial CRE isolation. The proposed manual action relies upon redundant, safety related control switches for CRPAR initiation and CRE isolation prior to movement of recently irradiated fuel assemblies. The functionality of the CRPAR system is not changed and the safety function is enhanced by complete CRE pre-isolation.*

- Could create "special circumstances" under which compliance with existing regulations may not produce the intended or expected level of safety and plant operation may pose an undue risk to public health and safety.

  Response: *No special circumstances are present in this application. There is no substantial increase in the likelihood or consequences of accidents that are beyond the design and licensing basis for KPS. There is no change in the levels of defense or cornerstones of reactor safety with this application. The proposed change does not significantly reduce the availability, or reliability of structures, systems, components or other human actions that are risk significant but are not required by regulations. Finally, the proposed change does not involve a change for which synergistic or cumulative effects could significantly impact risk.*

  The proposed HAs are simple and effective and are not subject to potential impacts of "special circumstances."

2. <u>Assess the safety significance of the HAs</u>

NUREG-1764 discusses two methods for determining the safety-significance of HAs. The first method, the Estimated Importance Method, requires an estimate of the risk-importance of the HA. The second, the Generic HA Method is based upon general risk information and some plant-specific information.

DEK has performed a safety significance review based on the Estimated Importance Method. This method was selected as the most appropriate based on the limited information in the KPS PRA regarding the proposed operator actions. This assessment is as follows:

<u>Estimated Importance Method - Preliminary Screening</u>

The Locked Rotor Accident (LRA) is a Condition IV event, which means it is not expected to occur during the life of the plant. It is included in the "General Transient" category in the KPS PRA, which is standard practice among U.S. plants. The LRA is a core damage concern only if the reactor fails to trip. However, fuel damage is assumed in the LRA even with a reactor trip. The LRA with successful reactor trip is documented in the USAR as affecting only a small portion of the core. Note: The USAR Section 14 LRA does credit reactor trip on low flow AND assumes some fuel failure up to 25% of the fuel rods. Core damage for probabilistic risk assessment purposes is defined as "uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects" (RG 1.200). Since an LRA causes damage to only a fraction of the core rather than the majority of the core (i.e. complete core uncovery), it does not meet that definition. The combined probability of an LRA with a failure of the reactor to trip is low enough that it is not modeled in probabilistic risk assessments (PRAs). Therefore, the operator action to initiate safety injection after an LRA is not modeled in the PRA and is preliminarily screened as Level III.

The Fuel Handling Accident (FHA) results in clad damage to only one fuel assembly. Since a FHA results in only clad damage rather than fuel damage and affects only one fuel assembly rather than the entire core, it does not meet the definition of core damage. Therefore, the operator action to place control room ventilation in post-accident recirculation mode after a FHA is not modeled in the PRA and is preliminarily screened as Level III.

3. Qualitatively assess the safety-significance of HAs

Three types of qualitative assessment are used:

a. Personnel Functions and Tasks
b. Design Support for Task Performance
c. Performance Shaping Factors

Three types of assessments are discussed as follows:

a. Personnel Functions and Tasks

This type of qualitative assessment examines the potential effects of the proposed HA for changes to operator tasks and the functions that they perform, under five major categories:

- Operating Experience: Does the requested change adversely affect the performance of an action that was previously identified as problematic based on experience/events at that plant or plants of similar design?

Response: *No, the requested change does not adversely affect the performance of an action that was previously identified as problematic. Currently, there are no required actions associated with CRPAR performance or CRE isolation. Routine operator actions associated with CRPAR operation (verifying lights, verifying annunciators, damper position, etc.) are not impacted by the proposed manual operator actions. Manual initiation of SI during a LRA with fuel damage is already a required action to prevent loss of subcooling to the core. This manual action is not changed.*

- New Actions: Does the requested change introduce new HAs? Are the new HAs associated with new responsibilities for the success of safety functions (or additional actions associated with existing responsibilities)?

  Response: *Yes, the change does introduce new HAs. The proposed HAs are associated with success of the safety function for protection of the control room occupants. For the FHA, the safety function associated with the HA is initiation of at least one train of the CRPAR system. The proposed HA supports long-term operation of KPS during and after a FHA by ensuring that control room occupants are exposed to as low a radiological dose as possible. For the LRA, the safety function associated with the HA is manual initiation of SI, which causes isolation of the CRE and initiation of the CRPAR system. The proposed HA supports long-term operation of KPS during and after a LRA by ensuring that control room occupants are exposed to as low a radiological dose as possible.*

- Change in Automation: Has the requested change given personnel a new functional responsibility that they previously did not have and which differs from their normal responsibilities? For example, are operators now required to take an action in place of a previously automated one? Consider the example of simply being required to open a valve that previously was automatically operated, and where the action required to do so is similar to other valve-opening operations with which operators are familiar. This would not be a sufficient change (in and of itself) to warrant a "yes" to this question when considering task complexity. However, there may be increased workload if the aggregate of added actions is judged to be excessive, this may warrant a "yes."

  Response: *No, while new tasks are required as discussed above, the example in this question is directly applicable to the proposed HA. The HA for the FHA is simply to turn control room hand switches associated with CRPAR system. This act is similar to the example of opening a valve that was previously automated. Operators are familiar with this type of action and routinely perform similar actions. This action is not complex and would be associated only with a FHA. For the LRA with fuel damage, the proposed HA is to manually initiate push buttons associated with Safety Injection. This is an action already* *required for loss of subcooling, and is therefore not a new action to the operators. Therefore, the proposed HAs are not excessive and are considered a minor increase in the workload for these events.*

- Change in Tasks: Has the requested change significantly modified the way in which personnel perform their tasks (e.g., making them more complex, significantly reducing the time available to perform the action, increasing the operator workload, changing the operator role from primarily "verifier" to primarily "actor")? In this case, operators do not have a new functional responsibility; instead, the way that they perform their current functional responsibilities has significantly changed and is different from what they usually do.

  Response: *The proposed HAs do not significantly change the way in which operators perform their tasks. As described above, operators routinely monitor the control room indications and plant status during and after an event. Initiation of one train of the CRPAR system is not complex and is consistent with the operator's role during an event. In addition, for the LRA with fuel damage, an expected response of the operator is to manually initiate SI upon a loss of subcooling.*

  As discussed in Section 2.0, one train of the CRPAR system must be initiated within 20 minutes for the FHA event. The CRE is required by TS 3.7.10 to be isolated prior to moving recently irradiated fuel assemblies. As discussed in Section 1.0, at least 1 hour is available from the initiation of an LRA before the HA is required to be completed. The HA consists of manually depressing SI signal push buttons, which causes isolation the CRE and initiation of the CRPAR system. Once these HAs are complete, the control room occupants will be provided protection during the FHA and LRA. Therefore, there is no significant new functional responsibility or significant change in responsibilities for operators during these events.

- Change in Performance Context: Has the requested changed created, in some way, a new context for task performance? Or, does the change identify a previously unrecognized context? Or, does the request address a context previously not modeled or considered? If so, what are the important differences in context (e.g., different plant mode, plant behavior, timing of plant symptoms)?

  Response: *The proposed HAs will not create or modify the context for task performance. As described above, the proposed HAs would be performed after occurrence of a FHA or LRA event. The context of performing HAs during an accident scenario is a function that is required to be understood by operators in their training for accident response. Therefore, the context is expected and has not changed.*

b. Design Support for Task Performance

This type of qualitative assessment addresses how well the performance of the HAs is supported (e.g., with job aids):

- Change in Human-System Interfaces (HSIs): Has the requested change significantly changed the HSIs used by personnel to perform the task? For example, are personnel now performing their tasks at a computer terminal where previously they were performed at a control board with analog displays and controls?

  Response: *The proposed HAs would not change any HSIs. The proposed HAs require manipulation of controls that are known to the operators. No new controls or human-system interfaces are proposed. The proposed HAs are simple and routine for operators.*

- Change in Procedures: Has the requested change significantly changed the procedures that personnel use to perform the task, or is the task not supported by procedures?

  Response: *A significant change to the procedures that operators use to perform the proposed HAs is not necessary. For the LRA with fuel damage, plant emergency procedures currently require manual initiation of SI if a loss of subcooling occurs. For the FHA, the manipulation of CRPAR system switches will be directed by station operating procedures as part of implementation of this amendment.*

- Change in Training: Has the requested change significantly modified the training, or is the task not addressed in training?

  Response: *The proposed HAs have been provided to operators in training. For the LRA with fuel damage, plant emergency procedures currently require manual initiation of SI if a loss of subcooling occurs. Operator training requires the operator to memorize this step. For the FHA, the initiation of one train of CRPAR system is provided in training and the reasons/basis for performing this HA is discussed in training.*

c. Performance Shaping Factors

This type of qualitative assessment addresses four performance shaping factors:

- Changes in Teamwork: Has the requested change significantly changed the team aspects of performing an action. For example, (1) is one operator now performing the tasks accomplished by two or more operators in the past? (2) is it now more difficult to coordinate the actions of individual crew members? or, (3) is task performance more difficult to supervise after the modification?

Response: *No changes in teamwork are required. No additional operators are required in the control room to perform the proposed HAs. There is no greater level of difficulty and no increase in the level of supervision necessary to accomplish proposed HAs. Manipulation of control room switches is a routine evolution for operators, and no additional level of supervision is required.*

- Changes in Skill Level of Individuals Performing the Action: Has the requested change kept the same HA but made it necessary for an individual who is less trained and has lower qualifications to take the action than was the case before the modification? Here, context is defined as the overall performance environment, including plant conditions and behavior that, for example, affect the time available for the operator response and the effectiveness of job aids under these conditions that lead to the assessment of performance shaping factors.

  Response: *The skill level of the operator performing the proposed HAs and the performance environment for the operator will not change. For the LRA with fuel damage, the procedural requirement to initiate SI is required to be memorized by operators. This has not changed. For the FHA, initiation of one train of the CRPAR system via hand switches is a routine type task. Job aids consist of control switch identification placards on the control boards and understanding when the SI manual push buttons and CRPAR system switches need to be initiated. These are simple routine tasks for operators and do not require new skills or additional training to accomplish.*

- Change in Communication Demands: Has the requested change significantly increased the level of communication needed to perform the task? For example, must an operator now communicate with other personnel to perform actions that previously could be taken at a local panel containing all necessary HSIs?

  Response: *The proposed HAs do not require significantly increased levels of communication to accomplish. Direction communicated by the unit supervisor during an event is considered a routine communication. Accomplishment of the proposed HAs is easily verified by lights and annunciators in the control room. In addition, operators are accustomed to working in pairs for peer checking and independent verification of system alignments.*

- Change in Environmental Conditions: Has the requested change significantly increased the environmental challenges (such as radiation, or noise) that could negatively affect task performance?

Response: *No, the operator will be performing the proposed HAs to ensure the CRE does not become a high radiation environment. There is no change in noise level associated with the proposed manual actions as these are the dampers and systems that normally provide air to the control room.*

4. Make an integrated assessment of HA Safety-Significance

The results of the qualitative assessment of HA Safety Significance have determined that the action is well defined and can easily be performed (it is clear when to perform the action), procedural direction exists, there is sufficient time and staff available to perform the action, and the action is similar to those routinely performed. Based on this, the level of HF review could be reduced to Level III. The Level III classification is warranted since most of the areas reviewed were answered "no" and the analysis indicates very little change is being made.

However, since the action involves support of a safety function and failure to accomplish the proposed HAs could potentially result in the loss of a high risk component (loss of the control room, via high radiation dose, is a PRA initiating event), the level classification will conservatively remain at Level II.

3.2. Phase II - HFE Review of Proposed HA using Level II Review Criteria

Based on the results of the Phase I Risk Screening provided above, DEK has conservatively determined that the proposed HAs will be assessed using the Level II criteria identified in Section 4 of NUREG-1764. NUREG-1764 specifies that a Level II review include the following elements:

1. General Deterministic Review
2. Analysis
3. Design of Human System-Interfaces, Procedures and Training
4. Human Action Verification

These four elements are assessed below:

1. General Deterministic Review Criteria

   Objective: The objective of this section is to verify that deterministic aspects of design, as discussed in RG 1.174, have been appropriately considered by the licensee. Deterministic aspects include verifying that the change meets current regulations and does not compromise defense-in-depth.

   Scope: The deterministic review criteria are applicable to all modifications associated with Level II HAs.

   Criteria:

1) The licensee should provide adequate assurance that the change meets current regulations, except where specific exemptions are requested under 10 CFR 50.12 or 10 CFR 2.802. Examples of regulations that may be affected by a change, but that may be identified as risk-significant when using a standard PRA to screen for risk include the following: 10 CFR Part 20, Criterion 19 of Appendix A to 10 CFR Part 50, and Appendices C through R to 10 CFR Part 50.

   Response: *See discussions located in Attachment 1 and Attachment 4 related to compliance with regulations and conformance to accident analyses criteria. See section 5.2 for a discussion of compliance with General Design Criteria applicable to KPS.*

2) The licensee should provide adequate assurance that the change does not compromise defense-in-depth:

   Response: *Defense-in-depth is one of the fundamental principles upon which KPS was designed and built. Defense-in-depth uses multiple means to accomplish safety functions and to prevent the release of radioactive materials. It is important in accounting for uncertainties in equipment and human performance, and for ensuring some protection remains even in the face of significant breakdowns in particular areas.*

   Defense-in-depth is not compromised or altered as a result of the proposed HAs. Defense-in-depth is accomplished in this particular case by having multiple reliable methods to contain highly radioactive materials during a design bases accident. The containment structure, shield building ventilation system, and auxiliary building special ventilation system all minimize the release or act upon the release of highly radioactive materials should barriers fail. Each of these systems has the goal of protecting the health and safety of the public and the control room occupants during an event. The proposed HAs ensure that the CRE is isolated and a filtered source of air is available for the control room occupants. This function is not compromised by performance of the proposed HAs.

   The proposed HAs do not lead to an over-reliance on programmatic activities to compensate for weaknesses in plant design. System redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties (e.g., no risk outliers).

   The proposed HAs preserve defenses against potential common cause failures, and there is no potential for the introduction of a new common cause failure mechanism. The proposed HAs include initiation of an SI signal which causes *actuation of the CRPAR system. Therefore, the independence of barriers is not compromised. Defenses against human errors are preserved because the proposed HAs are included in procedures and are included in operator training. Human errors, should they occur, are easily detectable by control room annunciators and equipment status lights.*

2. Analysis

Objective: The objective of the review is to verify that the licensee has analyzed the changes to HAs and identified HFE inputs for any modifications to the HSI, procedures, and training that may be necessary.

Scope: The review criteria are applicable to all modifications associated with Level II HAs.

Criteria:

1) Functional and Task Analysis

The licensee should identify how the personnel will know when the HA is necessary, that it is performed correctly, and when it can be terminated.

Response: *The need for performing the proposed HA associated with the LRA will be identified by multiple alarm conditions. The following conditions indicate the need for operator action during a LRA event (details are provided in section 3.6.1 above):*

- *A sudden decrease in core coolant flow which results in fuel damage as indicated by the RCS subcooling monitor (loss of sufficient cooling to the fuel).*
- *Upon indication of loss of subcooling, operators enter emergency operations procedures and initiate SI based on loss of subcooling.*
- *Initiation of an SI signal causes isolation of CRE dampers and initiation of the CRPAR system.*

Indication of correct performance of this proposed HA is made by verifying that the train of SI selected to respond to the LRA is functioning by observing that the indicating lights associated with the selected control switch change color from green (Standby) to red (On). In addition, the train of CRPAR automatically selected by association with the selected SI train for the mitigation of a LRA is verified to be functioning by observing that indicating lights associated with the selected control switch change color from green (Standby) to red (On) and verifying that the associated CRAC fan automatically starts by observing the indicator light for this fan is red (On). Finally, the CRE boundary is verified to *be intact by observing indication of damper positions in the closed position. This can be done by observing indications on damper control switches.*

Termination of CRPAR operation is not necessary until radiation conditions return to normal background levels, indicating that a radioactive release is no longer occurring.

The following conditions indicate the need for operator action during a FHA:

- *Verbal communication from personnel on the containment operating floor or spent fuel pool area that indicates a FHA has occurred.*
- *High radiation alarm indicated on R-2, R-5, R-12 or R-21 indicates a FHA in containment or the spent fuel pool area has occurred.*

Indication of correct performance of the proposed HA is made by verifying that the train of CRPAR selected for the mitigation of a FHA is functioning. This is done by observing that the indicating lights associated with the selected control switch change color from green (Standby) to red (On) and verifying that the associated CRAC fan automatically starts by observing the indicating light is red (On). The CRE boundary would have previously been verified to be intact by observing indication of damper positions in the closed position prior to moving recently irradiated fuel.

Termination of CRPAR operation is not necessary until radiological conditions return to normal background levels, indicating that a radioactive release is no longer occurring.

Task analyses should provide a description of what the personnel must do. The licensee should identify how human tasks or performance requirements are being changed. The task analysis should identify reasonable or credible, potential errors and their consequences.

Response: *Refer to Sections 1.0 and 2.0 for a detailed description of the proposed HAs. The proposed HAs are only required for the LRA and the FHA.*

<u>LRA</u>

There are a limited number of credible failures or errors that the operator can make during the LRA event. As shown above, proposed HA is not complex *(only requires manipulation of the manual SI push buttons) and requires little effort to complete. However, human errors do occur and the proposed HA accounts for the consequences of such errors. For example, if an operator fails to initiate SI, then multiple alarms indicating high radiation conditions may occur at various locations (e.g. R-9, R-11, R-12, or R-21). If the SI signal fails to isolate the control room, then the alarm from control room radiation monitor R-1 will not clear without some action being taken. As a backup, R-23 (although not credited in the radiological accident analyses) is still functional and will actuate on a high radiation condition. Similarly, if the SI signal does not start the associated train of CRPAR, then a high radiation condition will continue to persist until the control room operator manually starts a train.*

FHA

There are a limited number of credible failures or errors that the operator can make during the manipulations. As shown above, the proposed HA is not complex and requires little effort to complete. However, human errors do occur and the proposed HA accounts for the consequences of such errors. For example, if an operator fails initiate the CRPAR system, then the alarm from radiation monitor R-1 will not clear without some action being taken. As a backup, R-23 (although not credited in the analyses is still functional) will actuate on a high radiation condition.

If the operator incorrectly manipulates the wrong control room switches, the most likely outcome will be that the CRPAR system will not be initiated and high radiation alarms in the control room would continue. This would alert the operator that the incorrect switch was manipulated and could be corrected immediately.

2) Staffing:

The effects of the changes in HAs upon the number and qualifications of current staffing levels of operations personnel for normal and minimal staffing conditions.

Response: *is the proposed HAs would have no effect on the number and qualifications of operations personnel required to support operations in a post-event condition. It is routine for operators to monitor control room conditions and verify proper operation of equipment in the control room.*

3. Design of Human System-Interfaces, Procedures, and Training

Objective: The objective of the review is to verify that the licensee has supported the HAs by appropriate modifications to the HSI, procedures, and training.

Scope: The review criteria are applicable to all modifications associated with Level II HAs.

Criteria:

1) HSIs:

Temporary and permanent modifications to the HSI should be identified and described. The modifications should be based on task requirements, HFE guidelines, and resolution of any known operating experience issues.

Response: *No HSI modifications or new HSIs are required. The proposed HAs are simple and control room switches and push buttons are well marked.*

2) Procedures:

Temporary and permanent modifications to plant procedures should be identified and described. The modifications should be based on task requirements and resolution of any known operating experience issues. Justification should be provided when the plant procedures are not modified for changes in operator tasks.

Response: The appropriate modifications to plant procedures will be made as part of the implementation of this amendment request.

3) Training:

Temporary and permanent modifications to the operator training program should be identified and described. The modifications should be based on task requirements and resolution of operating experience issues. Justification should be provided when the training program is not modified for changes in operator tasks.

Response: *Training lesson plans will be revised to incorporate the bases for performing the proposed HAs contingent upon approval of this amendment request. The requirements of the training will be developed using the process specified in DEK training development procedures.*

4. <u>Human Action Verification</u>

Objective: The objective of this review is to verify that the licensee has demonstrated that the HAs can be successfully accomplished with the modified HSI, procedures, and training.

Scope: The review criteria are applicable to all modifications associated with Level II HAs.

Criteria:

1) An evaluation should be conducted at the actual HSI to determine that all required HSI components, as identified by the task analysis, are available and accessible.

   Response: *DEK has performed a walkdown of the control room and has verified that components required to perform the proposed HAs are accessible and available to the operator.*

2) A walkthrough of the HAs under realistic conditions should be performed to determine that;

   - The procedures are complete, technically accurate, and usable.
   - The training program appropriately addressed the changes in plant systems and HAs.

   The HAs can be completed within the time criterion for each scenario that is applicable to the HAs. The scenario used should include any complicating factors that are expected to affect the crews' ability to perform the HAs.

   Response: *As part of the walkdown described above, DEK developed and verified the procedures to be used as guidance to the operators for performing the proposed HAs. The procedures are used during the training of the operators to achieve a simulated performance of operator actions during training sessions.*

3) The walkthroughs should include at least one crew of actual operators.

   Response: *Operations personnel were included in the walkdown of the control room.*

4.0 Conclusions

This evaluation has demonstrated that the proposed HAs are acceptable. The Phase I Risk Screening (Section 3.1) demonstrated that the safety significance of the proposed HAs is minimal and warranted only a Level III Human Factor review. However, since the proposed HAs involved action that support a safety function, and failure to perform the proposed HAs could potentially result in the loss of a high risk component (a PRA initiating event), the classification level was conservatively left at Level II for the purposes of reviewing the HAs.

The results of the HFE review of the proposed HAs have determined that the four elements of the HFE review have been satisfied without identifying any obstacles to implementation. All the HFE elements of a Level II review were satisfied including:

- The technical review provides adequate assurance that the proposed HAs meet current regulations.
- The proposed HAs have been analyzed for their impact on current procedures, control room staffing, human system interfaces, and training.
- The proposed HAs are captured in procedures and in training.
- It has been demonstrated that the proposed HAs can be accomplished with the procedures and training provided.

5.0 References

1. RIS 2005-20, Revision 1, "Revision to NRC Inspection Manual Part 9900 Technical Guidance, Operability Determination & Functional Assessments for Resolution of Degraded and Nonconforming Conditions Adverse to Quality or Safety," dated April 16, 2008.
2. Information Notice 97-78, "Crediting of Operator Actions in Place of Automatic Actions and Modifications of Operator Actions, Including Response Times," dated October 23, 1997.
3. NUREG-1764, Revision 1, "Guidance for the Review of Changes to Human Actions," dated January 2005.
4. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002. [ADAMS Accession No. ML023240437]
5. NUREG-0800, "Standard Review Plan," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant Specific Changes to the Licensing Basis: General Guidance," dated June 2007. [ADAMS Accession No. ML071700658]