March 8, 2007

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION - ISSUANCE OF AMENDMENT RE: RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE (TAC NO. MC9715)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 190 to Facility Operating License No. DPR-43 for the Kewaunee Power Station. This amendment revises the Technical Specifications in response to your application dated January 30, 2006, as supplemented by letter dated January 23, 2007.

The amendment revises radialogical accident analyses and associated technical specifications.

A copy of the NRC staff's Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular biweekly *Federal Register* notice.

Sincerely,

/RA/

Robert F. Kuntz, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 190 to
   License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

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DATE	3/07/07	3/06/07	2 /22/07	7/21/07	2/28/07	3/08/07

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Kewaunee Power Station

cc:

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DOMINION ENERGY KEWAUNEE, INC.

DOCKET NO. 50-305

KEWAUNEE POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 190
License No. DPR-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Dominion Energy Kewaunee, Inc. dated January 30, 2006 as supplemented by letter dated January 23, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Lakshminaras Raghavan, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: March 8, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 190

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Replace the following page of the Facility Operating License No. DPR-43 with the attached revised page. The changed area is identified by a marginal line.

REMOVE	INSERT
Page 3	Page 3

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
TS 3.1-7	TS 3.1-7
TS 3.6-4	TS 3.6-4
TS 6.20-1	TS 6.20-1

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: (1) Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, (2) is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and (3) is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 190, are hereby incorporated in the license. The licensee shall operate the facility In accordance with the Technical Specifications.

(3) Fire Protection

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

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

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21, is entitled: "Nuclear Management Company Kewaunee Nuclear Power Plant Physical Security Plan (Revision 0)" submitted by letter dated October 18, as supplemented by letter dated October 21, 2004, July 26, 2005, and May 15, 2006.

(5) Fuel Burn-up

The maximum rod average Burn-up for any rod shall be limited to 60 GWD/MTU until completion of an NRC environmental assessment supporting an increased limit.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 190 TO FACILITY OPERATING LICENSE NO. DPR-43

DOMINION ENERGY KEWAUNEE, INC.

KEWAUNEE POWER STATION

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated January 30, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML060540217), as supplemented by letter dated January 23, 2007 (ADAMS Accession No. ML070240543) Dominion Energy Kewaunee (DEK, the licensee) requested an amendment to facility Operating License No. DPR-43 for the Kewaunee Power Station (KPS). The licensee proposed to modify the currently approved radiological accident analyses and associated technical specifications (TSs). This proposed amendment incorporates TS changes to compensate for the higher control room emergency zone (CREZ) unfiltered inleakage measured during the American Society for Testing and Materials (ASTM) E741 (tracer gas) leakage test conducted in December 2004.

Results from the ASTM testing of the KPS control room envelope showed the CREZ unfiltered inleakage to be greater than that assumed in the approved radiological accident analyses. The revised CREZ unfiltered inleakage was determined to be a facility change, which caused an increase in the dose consequences of the approved radiological accident analyses. Currently, the KPS CREZ is operable but non-conforming. The resolution of this condition is to incorporate the increase in assumed CREZ unfiltered inleakage into the radiological accident analyses.

The supplemental letter contained clarifying information, did not change the initial no significant hazards consideration determination, and did not expand the scope of the original *Federal Register* notice.

The licensee proposed changes to the TSs for certain plant parameters to compensate for the higher measured CREZ unfiltered inleakage. This safety evaluation (SE) addresses the Nuclear Regulatory Commission (NRC) staff's review of the licensee's revised radiological accident analyses. In this license amendment request, the licensee proposed to change:

1. TS 3.1.c.2.A, "Maximum Coolant Activity," coolant activity limit that requires intermediate shutdown from 60 μCi/gram DOSE EQUIVALENT I-131 to 20 μCi/gram DOSE EQUIVALENT I-131.

2. TS 3.6.c.3.B, “Performance Requirement,” Shield Building Ventilation System and the Auxiliary Building Special Ventilation system filter removal efficiency from $\geq$95 percent radioactive methyl iodide removal to $\geq$97.5 percent radioactive methyl iodide removal.

3. TS 6.20, “Containment Leakage Rate Testing Program,” maximum allowable leakage rate from 0.5 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig to 0.2 weight percent.

2.0 REGULATORY EVALUATION

The U.S. Nuclear Regulatory Commission (NRC) staff finds that the licensee in Section 5.2 of its January 30, 2006, submittal, identified the applicable regulatory requirements. The regulatory requirements and guidance which the staff considered in its review of the requested action are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion 19 (GDC-19) requires, in part, that “[a] control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.”

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36 “Technical specifications,” requires that “an applicant for a license authorizing operation of a production or utilization facility include proposed technical specifications” in its license application.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67 “Accident source term” establishes analyzed dose limits for acceptable adoption of the accident source term. Title 10 of the *Code of Federal Regulations* (10 CFR) 50.67(b)(2) states that “[t]he NRC may issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem)[22] total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

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[2]The use of 0.25 SV (25rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 SV (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident."

The NRC staff also considered the guidance contained in Regulatory Guide (RG) 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors" as well as NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) Section 15.0.1"Radiological Consequence Analyses Using Alternative Source Terms" in reviewing the licensee's amendment request.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee's analysis methods, assumptions, and inputs using docketed information provided by the licensee.

These radiological accident analyses of record for the KPS license were previously docketed in Amendment No. 166, issued March 17, 2003 (ADAMS Accession No. ML030210062), which implemented an alternate source term; and Amendment No. 172, issued February 27, 2004 (ADAMS Accession No. ML040430633), which implemented a stretch power uprate to 1772 mega-watt thermal (MWt). These previously approved radiological accident analyses used the analytical methods and assumptions outlined in RG 1.183.

The revised radiological accident analyses for design-basis accidents (DBA) incorporate changes for the control room isolation parameters based on air flow measurements of ASTM E741 (tracer gas) testing conducted in response to Generic Letter 2003-01.

The control room envelope unfiltered inleakage was measured by tracer gas testing on December 14 and 15, 2004. The leak test measured 409 ± 29 cubic feet per minute (cfm) for system train A, and 447 ± 51 cfm for train B. In the revised radiological accident analyses, DEK adjusted the control room unfiltered inleakage rate during normal mode heating ventilation and air conditioning (HVAC) operation to range between 1620 to 2750 cubic feet per minute (cfm). DEK specified two unfiltered inleakage for control room emergency ventilation. For events that model isolation actuated by a safety injection (SI) set point, CREZ unfiltered inleakage was assumed to be at least 800 cfm. For events actuated by high radiation monitor in the control room air supply duct, the CREZ unfiltered inleakage was assumed to be at least 1500 cfm. Isolation actuated by the control room air supply duct radiation monitor does not close all control room isolation dampers. The unfiltered inleakage rate of 1500 cfm compensates for the dampers that remain open. DEK also increased the assumed control room damper closure time to 20 seconds from the previous value of 10 seconds. The increased damper closure time bounds actual measured closure times.

DEK compensated for the higher control room unfiltered inleakage in the radiological accident analysis by proposing modifications to TSs limits. The three proposed TS changes reduce the calculated fission product released to the environment, thus allowing for higher control room unfiltered inleakage. In this amendment request, DEK submitted revised radiological analyses of the DBAs. The dose acceptance criteria for the DBAs and the revised licensee-calculated radiological consequence are listed in Table 1. The CREZ unfiltered inleakage parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 2.

3.1 Main Steamline Break (MSLB) Accident

In the revised radiological analysis for MSLB accident, DEK changed the assumed CREZ unfiltered inleakage from 200 to 1000 cfm, which bounds the ASTM E741 test results. This is the only assumption that has changed from the previous radiological accident analysis of MSLB.

The licensee assumed that the faulted steam generator (SG) boils dry within 2 minutes. The entire liquid inventory of the faulted SG is steamed off and all the iodine initially in the SG is released to the outside environment. The primary-to-secondary SG tube leakage rate is assumed to be at the TS limit of 150 gallons per day (gpd) per SG. The 150 gpd leakage for the faulted SG, along with its noble gas and iodine, is assumed released directly to the outside atmosphere. In the intact SG, the 150 gpd primary-to-secondary leakage mixes with the bulk SG secondary coolant water. Transferred noble gases are released without holdup, and iodine is released to the outside environment at the steaming rate of the intact SG, with credit for partitioning when the SG tubes are covered with water.

DEK analyzed the MSLB for two iodine spiking cases. The pre-accident iodine spiking case assumed that a reactor transient has occurred prior to the MSLB, and has raised the reactor coolant system (RCS) iodine concentration to 60 micro curies per gram (µCi/gm) of dose equivalent (DE) Iodine 131 (I-131). The accident-initiated iodine spiking case assumed that the reactor trip associated with the MSLB creates an increase in the iodine release rate from the fuel to the coolant to a value 500 times greater than the release rate corresponding to a maximum equilibrium RCS concentration of 1.0 µCi/gm DE I-131. The accident-initiated spike duration is 4 hours. The secondary coolant activity in both cases is assumed to be the TS limit of 0.1 µCi/gm DE I-131. No fuel damage is projected for the MSLB.

The low steamline pressure SI set point will be reached shortly after the onset of an MSLB. The SI signal causes the control room HVAC to switch from normal operation mode to the accident mode of operation. DEK conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins.

The NRC staff reviewed the licensee's analysis of the MSLB radiological consequences, and finds that they remain consistent with the guidance provided in RG 1.183. The licensee's calculated radiological consequences at the Exclusion Area Boundary (EAB), Low Population Zone (LPZ) and in the KPS control room are within the dose criteria specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the MSLB. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 3) acceptable.

3.2 Locked Rotor Accident

In the revised locked rotor accident analysis, DEK changed the assumption for the fraction of failed fuel rods from 100 percent down to 50 percent, which is less conservative. DEK based the 50 percent assumption on the reload safety analysis limit. DEK also increased the length of time assumed for control room HVAC to enter accident mode of operation from 10 to 45 minutes. DEK also conservatively revised the CREZ unfiltered inleakage to 1500 cfm based on tracer gas test results, and because the control room air supply duct radiation monitor actuates the HVAC accident mode of operation. These are the only assumptions that have changed from the previous radiological accident analysis of the locked rotor accident.

The licensee's analysis assumes that a reactor transient has occurred prior to the locked rotor accident and that the transient has raised the RCS iodine activity concentration to 60 μCi/gm DE I-131, which bounds the proposed TS 3.1.c.2.A limit of 20 μCi/gm DE I-131. The noble gas and alkali metal activity concentration in the primary coolant is based on a fuel defect level of 1.0 percent. The iodine activity concentration in the secondary coolant is assumed to be 0.1 μCi/gm DE I-131, and the alkali metal activity concentration is assumed to be 10 percent of the primary coolant concentration. Accident-induced activity is assumed to be released to the environment as a result of primary-to-secondary leakage through the SG tubes and steaming from the secondary side, released through either the atmospheric relief valves or safety valves. An iodine partitioning factor in the SGs of 0.01 is used to account for retention of iodine in the SG as the water turns to steam. The partitioning factor of 0.01 is also applied to the alkali metal activity release. All noble gas activity carried over to the secondary side of the SGs is assumed to be immediately released to the outside atmosphere. At 8 hours after the accident, the licensee assumed that the residual heat removal (RHR) system has removed all decay heat with no further releases to the environment after that time.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the locked rotor accident and finds that they are consistent with the guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the locked rotor accident. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 4) acceptable.

3.3 Control Rod Ejection Accident

In the revised radiological analysis of the control rod ejection accident, DEK maintained the same assumptions as applied in previous radiological analysis, except the CREZ unfiltered inleakage rate has been conservatively increased to 1000 cfm to account for tracer gas test results.

This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assemble and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melting are projected. The radioactivity in the primary coolant is assumed to leak through the SG tubes into the secondary coolant. A portion of this activity is released to the outside atmosphere through the main condenser, atmospheric relief valves or safety valves. Additionally, radioactive primary coolant is discharged to the containment through the opening in the reactor vessel head where the control rod assembly was ejected. The activity in the containment is assumed to be released to the environment as a result of design-basis containment leakage evaluated at the proposed TS limit of 0.5 percent per day for the first 24 hours. After that, the containment is assumed to leak at half that rate until the end of the 30-day period considered in the analysis. In each case, the containment and secondary coolant release pathways are considered separately with bounding source term release for the combined release path ways.

DEK assumed that 15 percent of the fuel rods in the core suffer sufficient damage such that all their gap activity is released. The licensee assumed that 10 percent of the total core activity of iodine and noble gases and 12 percent of the total core activity for alkali metals are in the fuel gap, consistent with guidance provided in RG 1.183. A small fraction of the fuel in the failed rods is assumed to melt as a result of the rod ejection. The licensee estimated this melting to be limited to 0.375 percent of the core. This estimate was previously accepted in amendment No.166 (ML030210062).

The licensee assumed 100 percent of noble gases and alkali metals in the failed fuel gap and melted fuel are released to either the RCS or the containment, depending on the pathway assumed. For the containment leakage pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 25 percent of the iodine released from melted fuel are released to the containment atmosphere. For the primary-to-secondary leakage release pathway, the licensee assumed that all the iodine from the gap of the failed fuel and 50 percent of the iodine released from melted fuel is released to the RCS.

As discussed for the locked rotor accident (Section 3.2 of this SE), the licensee assumed an iodine partitioning factor of 0.01 in the SGs for the primary-to-secondary leakage release pathway. For the containment leakage release pathway, no credit was taken for iodine or particulate removal mechanisms.

The low pressurizer pressure SI setpoint is expected to be reached within 60 seconds of the onset of the control rod ejection. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. For this accident, DEK conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2.5 minutes after the event begins.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the control rod ejection accident and finds that they remain consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the control rod ejection accident. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 5) acceptable. To verify the licensee's dose results the NRC staff performed confirmatory calculations for the control rod ejection accident and finds the licensee's results to be reasonable.

3.4 Steam Generator Tube Rupture (SGTR)

In the revised radiological analysis of the SGTR, DEK changed the assumed pre-accident iodine spike from a value of 60 μCi/gm DE I-131 to a value of 20 μCi/gm DE I-131 per TS 3.1.c.2.A. DEK also increased the CREZ unfiltered inleakage from 200 to 1000 cfm based on the tracer gas test results.

The SGTR is analyzed for two iodine spiking cases; a pre-existing iodine spike resulting in elevated primary coolant activity, and an iodine spike initiated by the accident. For the pre-existing iodine spike case, the RCS iodine activity concentration is assumed to be at the proposed TS 3.1.c.2.A limit for a transient, equal to 20 µCi/gm DE I-131. For the accident initiated iodine spike case, the associated reactor trip causes an increase in the iodine release rate from the fuel to the RCS to a value 500 times the rate associated with the TS equilibrium RCS activity concentration of 1.0 µCi/gm DE I-131. The duration of the accident initiated iodine spike is limited by the amount of iodine in the fuel gap. Based on having 8 percent of the core inventory of iodine in the fuel gap, the spike would last 4 hours. RG 1.183 allows an accident initiated spiking factor of 335 for the SGTR, and the NRC staff finds the licensee's assumed factor of 500 is conservative compared to the RG value. All other analysis inputs are consistent with the guidance in RG 1.183.

The low pressurizer pressure SI setpoint is expected to be reached at around 2.9 minutes after the onset of the SGTR. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 5 minutes after the event begins.

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the SGTR and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room, are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the SGTR. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 6) acceptable.

3.5 Large-Break Loss-of-Coolant Accident (LBLOCA)

For the LBLOCA analysis, the current radioactive methyl iodide removal percentage TS limit for the shield building ventilation system, and the Auxiliary Building Special Ventilation System carbon filters is ≥ 95 percent. The licensee reviewed historical data of radiological accident analysis (RAA) sensitivity cases, to ensure that although more limiting, the proposed conservative change to ≥ 97.5 percent is reasonable and continues to provide adequate operating margin. The NRC staff finds that the revised limits bound plant charcoal filter test results and provide sufficient operating margin and are therefore acceptable.

For the LBLOCA analysis, the containment leakage rate is reduced to 0.2 weight percent per day of the contained air per 24 hours at a peak test pressure of 46 psig from the current analysis value of 0.5 weight percent per day. The licensee reviewed historical data of RAA sensitivity cases, to ensure that although more limiting, the proposed conservative change is reasonable and continues to provide adequate margin to actual measurements of containment leakage rates. The NRC staff finds that the revised containment leak rate limit bounds the plant measured containment leak rate test result, and provides sufficient operating margin and is therefore acceptable.

In its revised radiological analysis of LBLOCA, DEK changed the assumed shield building and auxiliary building filter efficiencies, the containment leakage rate, and the CREZ unfiltered inleakage flow rate. The revised assumptions increased shield building and auxiliary building filter efficiencies from 90 percent to 95 percent for removal of both elemental and organic iodine. The containment leak rates are revised from 0.5 to 0.2 weight percent per day for the first 24 hours, per proposed TS 6.20, and from 0.25 to 0.1 weight percent per day for greater than 24 hours. The CREZ unfiltered inleakage flow rate is revised from 200 to 800 cfm based on tracer gas test results. These are the only assumptions that have changed from the previous radiological accident analysis of LBLOCA.

In the licensee's analysis of the LBLOCA radiological consequences, activity from the damaged core is released into the containment. Three pathways for release to the environment are considered in the analysis:

(1) design-basis containment leakage,
(2) leakage from engineering safety feature (ESF) systems outside containment, and
(3) emergency core cooling system (ECCS) recirculation back-leakage to the refueling water storage tank (RWST).

The calculated radiological consequences of these three release pathways are added together to determine the total LBLOCA radiological consequences.

3.5.1 Containment Leakage Pathways

The containment is assumed to leak at the proposed TS 6.20 design-basis leak rate of 0.2 percent per day for the first 24 hours of the accident, and then to leak at half that rate for the remainder of the 30-day analysis period. The licensee assumed that during the first 10 minutes of the accident, 90 percent of the activity leaking from the containment is discharged directly to the environment. The remaining 10 percent enters the auxiliary building where it is released through filters. After 10 minutes, only 1 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 go into the shield building. The air discharged from the shield building is filtered. Additionally, once the shield building is brought to sub-atmospheric pressure at 30 minutes into the accident, iodine and particulate can be removed by recirculation through filters. A shield building participation fraction of 0.5 is assumed.

The shield building filter efficiency for elemental and organic iodine is revised to 95 percent, which is bound by the proposed level in TS 3.6.c.3.B. The licensee assumed removal of iodine through sedimentation for particulate and the containment spray for elemental and particulate forms of iodine. The KPS containment spray system is an ESF system and is designed to provide containment cooling and fission product removal in the containment following a LBLOCA. One train of spray was assumed to operate. Switch over to recirculation spray is not credited and all spray removal is terminated when the RWST drains down at 0.91 hours from the start of the accident. In determining the core spray iodine removal rates, the licensee assumed a reduction in assumed spray flow relative to that assumed in the analysis supporting Amendment No. 166 (ML030210062). This reduction is intended to bound potential pump degradation. The NRC staff finds this change acceptable. The licensee assumed a sedimentation coefficient of 0.1 hr^{-1} for particulate after the core spray system is terminated.

The licensee used the models and guidance provided in RG 1.183 and SRP 6.5.2, "Containment Spray as a Fission Product Cleanup System," to determine the removal rates for iodine.

3.5.2 Post-LOCA ESF Leakage Pathway

During the recirculation phase of long-term core cooling, radioactive water from the containment sump is sent to ECCS equipment located outside the containment. These components may leak into the auxiliary building. Although ECCS recirculation does not occur until 0.91 hours after the accident begins, the licensee conservatively assumed leakage occurs immediately upon the onset of the LBLOCA. The licensee conservatively assumed the leakage to the auxiliary building is 12 gallons per hour. The licensee assumed that 10 percent of the activity in the leaked fluid becomes airborne when the sump temperature is above 212 °F. Once the sump temperature drops below 212 °F at 3 hours from the start of the event, the airborne activity fraction is reduced to 1 percent of the activity in the leaked fluid. The assumed time that the sump temperature falls below 212 °F is selected to bound the results of the containment response analyses performed for the TS changes. The NRC staff finds these assumptions are consistent with RG 1.183 and are acceptable. The licensee also assumed that half of the airborne iodine activity in the auxiliary building is removed by plateout on surfaces. This assumption was previously approved in Amendment No. 166 (ML030210062).

3.5.3 ECCS Back-Leakage to the RWST

RHR back-leakage to the RWST is assumed to be at a rate of 3 gallons per minute (gpm) for the first 24 hours, and 1.5 gpm for the remainder of the accident. It is assumed that 1 percent of the iodine becomes airborne, even when the sump temperature is above 212 °F since any incoming water would be cooled by the water remaining in the RWST.

3.5.4 Control Room Ventilation System Modeling

For the LBLOCA, the low pressurizer pressure SI setpoint will be reached shortly after the start of the event. The SI signal causes the control room HVAC to switch from the normal operation mode to the accident mode of operation. The licensee conservatively assumed that the control room HVAC does not fully enter the accident mode of operation until 2 minutes after the event begins.

3.5.5 LBLOCA Conclusion

The NRC staff reviewed the licensee's methods, inputs and assumptions used in its revised radiological consequences analysis of the LBLOCA for TS changes, and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room are within the dose limits specified in 10 CFR 50.67 and GDC-19, and are within the acceptance criteria given in SRP 15.0.1 for the LBLOCA. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 7) acceptable. To verify the licensee's dose results, the NRC staff performed confirmatory radiological consequence analyses of the LBLOCA and finds the licensee's results to be reasonable.

3.6 Fuel-Handling Accident (FHA)

In the revised radiological analysis, DEK conservatively changed the CREZ unfiltered inleakage flow rate from 200 to 1500 cfm. This is the only assumption that has changed from the previous radiological accident analysis of FHA.

The licensee's analysis of the FHA was performed with assumptions selected so that the results are bounding for an accident that occurs either in the containment or in the auxiliary building. Activity released from the damaged assembly is assumed to be released to the environment through either the containment purge system or the spent fuel pool ventilation system, without credit for filtration or isolation of the containment, containment purge system, or spent fuel pool ventilation system. The decay time used, 100 hours, is the minimum decay time required by TS before movement of fuel. The licensee assumed that all the fuel rods in the equivalent of one fuel assembly are damaged, and all the gap activity in the rods is released to the pool. A pool iodine effective decontamination factor of 200 is assumed. All fuel gap noble gas activity is assumed to be released from the pool. All activity released from the pool is assumed to be released to the outside environment within 2 hours.

DEK assumed that the control room HVAC system is in normal operation mode at the onset of the FHA. A high-radiation signal for the control room air supply duct is generated as a result of the activity release to the atmosphere, and control room HVAC enters accident mode of operation within 25 minutes.

The NRC staff reviewed the licensee's methods, inputs and assumptions, and finds that they are consistent with the conservative guidance given in RG 1.183. The licensee's calculated radiological consequences at the EAB, LPZ and in the KPS control room, are within the dose limits specified in 10 CFR 50.67 and GDC-19 and are within the acceptance criteria given in SRP 15.0.1 for the FHA. The NRC staff finds the results of the licensee's calculations (Table 1), and the major parameters and assumptions used by the licensee (Table 8) acceptable. To verify the licensee's dose results, the NRC staff performed a confirmatory radiological consequence analysis of the FHA and finds the results to be reasonable.

3.7 Waste Gas Decay Tank (GDT) Rupture and Volume Control Tank (VCT) Rupture

The KPS licensing basis includes analyses of the radiological consequences of the waste GDT rupture, and the VCT rupture. The radiological analyses for these two accidents were previously found acceptable by the NRC staff in its SE approving Amendment No. 166 to the KPS license. The only change to the assumptions in the existing analyses is the increase in CREZ unfiltered inleakage. The NRC staff documented in its SE approving Amendment No. 172, issued February 27, 2004 (ADAMS Accession No. ML040430633) that "[b]ecause of the short duration of the radiation release, minimizing the assumed control room unfiltered inleakage maximizes the calculated control room dose. This is due to less dilution of the radioactivity in the control room. The NRC staff finds these assumptions to be acceptable based on plant operation and the operation of the radiation monitoring and control room HVAC systems." Therefore, the assumed increase in the unfiltered inleakage for these two accidents is bounded by the previous analyses. Consequently, these two accidents were not re-analyzed.

3.8 Technical Evaluation Conclusion

The NRC staff reviewed the assumptions, inputs, and methods used by DEK to reevaluate the radiological consequences of DBAs. The NRC staff finds that DEK used analysis methods and assumptions consistent with the conservative regulatory requirements and guidance identified in Section 2. DEK's analysis demonstrated that the radiological consequences of DBAs would remain within applicable regulatory limits. The NRC staff has performed confirmatory calculations on selected accidents and finds, with reasonable assurance, that the revised radiological accident analyses, assuming higher CREZ unfiltered inleakage and amended TS, (TS 3.1.c.2.A, TS3.6.c.3.B, and TS 6.20) complies with the regulatory requirements and is therefore acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (71 FR 13172). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

6.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: March 8, 2007

Table 1
Design-Basis Accident Licensee Calculated Radiological Consequences
TEDE (rem)

Design-Basis Accident	EAB	LPZ	Control Room
MSLB, Pre-existing iodine spike	0.03	0.01	0.70
Dose acceptance criteria	25	25	5
MSLB, Accident-initiated iodine spike	0.06	0.02	2.60
Dose acceptance criteria	2.5	2.5	5
Locked Rotor Accident	0.40	0.06	3.90
Dose acceptance criteria	2.5	2.5	5
Control Rod Ejection Accident	0.40	0.09	4.54
Dose acceptance criteria	6.3	6.3	5
SGTR, Pre-existing spiking	0.50	0.10	1.90
Dose acceptance criteria	25	25	5
SGTR, Accident-initiated spiking	0.80	0.20	2.80
Dose acceptance criteria	2.5	2.5	5
LBLOCA, total	0.52	0.09	4.95
Dose acceptance criteria	25	25	5
FHA	0.90	0.15	4.0
Dose acceptance criteria	6.3	6.3	5

Table 2
Revised Control Room Parameters

	REVISED ASSUMPTION	**PREVIOUS ASSUMPTION**
Normal ventilation flow rates		
Unfiltered Makeup Flow Rate	1620 - 2750 cfm	2250 - 2750 cfm
Emergency Ventilation Flow Rates		
Unfiltered Inleakage Following SI	$\geq$ 800 cfm	200 cfm
Unfiltered Inleakage Following R-23[3]	1500 cfm	200 cfm
Control Room Isolation Damper Closure Time	20 seconds	10 seconds

[3] The CREZ UFI is increased to at least 1500 cfm for events that model control room isolation on a control room radiation monitor, R-23, high control room duct activity monitor actuation (i.e. locked rotor and fuel handling accident).

Table 3
Assumptions Used in Radiological Consequence Analysis
Main Steamline Break

Reactor coolant activity	
Pre-existing iodine spike case	60.0 µCi/gm DE I-131
Accident-initiated iodine spike case	1.0 µCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant initial liquid mass	
Faulted steam generator (SG)	161,000 lbm
Intact SG	84,000 lbm
Steam release from faulted SG	161,000 lbm
Time to release faulted SG initial mass	2 minutes
Steam release from intact SG	
0 - 2 hours	222,000 lbm
2 - 8 hours	424,000 lbm
8 - 24 hours	614,000 lbm
Time to cool RCS and stop faulted SG release	72 hours
Steam partition coefficient	
Faulted steam generator	1
Intact steam generator	0.01
Steam generator tube leak rate	150 gallons per day per SG
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 4
Assumptions Used in Radiological Consequence Analysis
Locked Rotor Accident

Reactor coolant activity	60.0 μCi/gm DE I-131
Secondary coolant activity	0.1 μCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass 0 to 30 minutes Secondary coolant mass > 30 minutes	7.89E+07 gm 1.06E+08 gm
Secondary coolant mass > 30 minutes	1.06E+8 gm
Fuel rods in core failing, No fuel melting	50%
Peaking Factor Applied to Calculate Activity in Failed Fuel Rods	1.7
Fission product gap fractions	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Iodine chemical form in release	97% elemental, 3% organic
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 2 hours	210,000 lbm
2 - 8 hours	455,000 lbm
Steam partition coefficient	0.01
Time to cool RCS and stop steam release	8 hours
Time until begin control room emergency HVAC	45 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1500 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 5
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Reactor power	1782.6 MWt
Reactor coolant activity	60.0 µCi/gm DE I-131
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant mass	7.89E+07 gm
Radial peaking factor	1.7
Fuel rods in core failing	15%
Fission product gap fractions	
Iodines and noble gases	0.10
Alkali metals	0.12
Fuel rods in core melting	0.375%
Fission product activity released from melted fuel	
Noble gases and alkali metals	100%
Iodines	25% for containment leakage path 50% for SG steaming path
SG steaming release pathway	
Primary-to-secondary SG tube leak rate	150 gallons per day per SG
Steam release from secondary	
0 - 200 seconds	800 lbm/sec
200 - 1800 seconds	100 lbm/sec
> 1800 seconds	0 lbm/sec
Steam partition coefficient	0.01
Iodine chemical form in steam release	97% elemental, 3% organic
Containment leakage pathway	
Containment net free volume	1.32E+06 ft^3
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak rate	
0 - 24 hours	0.5 weight %/day
> 24 hours	0.25 weight %/day

Table 5 (continued)
Assumptions Used in Radiological Consequence Analysis
Control Rod Ejection Accident

Containment leak path fractions	
0 -10 minutes	
Through shield building	0.0
Through auxiliary building	0.1
Direct to environment	0.9
> 10 minutes	
Through shield building	0.89
Through auxiliary building	0.1
Direct to environment	0.01
Shield building air flow	
0 - 10 minutes	
Shield building to environment	Not applicable
Shield building recirculation	Not applicable
10 - 30 minutes	
Shield building to environment	6000 cfm (+10%)
Shield building recirculation	0.0 cfm
> 30 minutes	
Shield building to environment	3100 cfm
Shield building recirculation	2300cfm
Shield building and auxiliary building filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Time until begin control room emergency HVAC	2.5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 6
Assumptions Used in Radiological Consequence Analysis
Steam Generator Tube Rupture

Reactor coolant activity	
Pre-existing iodine spike case	20.0 µCi/gm DE I-131
Accident-initiated iodine spike case	1.0 µCi/gm DE I-131
Accident-initiated iodine appearance rate spiking factor	500 times equilibrium rate
Duration of accident-initiated iodine spike	4 hours
Secondary coolant activity	0.1 µCi/gm DE I-131
Primary coolant mass	1.19E+08 gm
Secondary coolant initial liquid mass	84,000 lbm/SG
Intact steam generator tube leak rate	150 gallons per day
Pre-trip releases (< 173.3 seconds)	
Tube rupture break flow	16,900 lbm
Percentage of break flow that flashes to steam	19.93%
Steam release to condenser	1077.8 lbm/sec for each SG
Post-trip releases (> 173.3 seconds)	
Tube rupture break flow	138,000 lbm
Percentage of break flow that flashes to steam	14.76%
Steam release from ruptured SG, 0 - 2 hours	86,400 lbm
Steam release from intact SG, 0 - 2 hours	233,400 lbm
Steam release from intact SG, 2 - 8 hours	488,800 lbm
Steam release from intact SG, 8 - 24 hours	662,800 lbm
Steam partition coefficient	
Ruptured steam generator, break flow	1
Intact steam generator	0.01
Time until begin control room emergency HVAC	5 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1000 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 7
Assumptions Used in Radiological Consequence Analysis
Large-Break Loss-of-Coolant Accident

Reactor power	1782.6 MWt
Source term	Based on RG 1.183
Containment volume	1.32E+06 ft^3
Containment leak rate	
0 - 24 hours	0.2 weight % per day
> 24 hours	0.1 weight % per day
Shield building volume	3.74E+05 ft^3
Shield building participation fraction	0.5
Containment leak modeling	See Table 4
Spray operation	
Time to initiate sprays	0.0 hours
Termination of sprays	0.91 hours
Recirculation spray	Not credited
Removal coefficients	
Elemental iodine	20 hr^{-1}
Particulate	4.5 hr^{-1}
Sedimentation (after spray termination)	0.1 hr^{-1}
ECCS leakage	
Containment sump volume	315,000 gal
ECCS leak rate, 0 - 30 days	12 gal/hr
Airborne percent iodine to auxiliary building	
0 - 3 hours	10%
> 3 hours	1%
ECCS leak rate to RWST	
0 - 24 hours	3 gpm
> 24 hours	1.5 gpm
Shield and auxiliary building filter efficiencies	
Elemental	95%
Organic	95%
Particulate	99%
Time until begin control room emergency HVAC	2 minutes
Normal ventilation flow rates	
Unfiltered makeup	2500 cfm (+10%)
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	800 cfm

Table 7 (continued)
Assumptions Used in Radiological Consequence Analysis
Large-Break Loss-of-Coolant Accident

Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 8
Assumptions Used in Radiological Consequence Analysis
Fuel-Handling Accident

Reactor power	1782.6 MWt
Radial peaking factor	1.7
Fission product decay period	100 hours
Number of fuel assemblies damaged	1
Fuel pool water depth	23 ft
Pool iodine effective decontamination factor	200
Fraction of fuel compliant with RG 1.183, footnote 11	0.50
Fuel gap fission product inventory (RG 1.183, footnote 11 compliant)	
I-131	8%
Kr-85	10%
Other iodines and noble gases	5%
Fraction of fuel not compliant with RG 1.183, footnote 11	0.50
Fuel gap fission product inventory (RG 1.183, footnote 11 non-compliant)	
I-131	12%
Kr-85	30%
Other iodines and noble gases	10%
Duration of release	2 hours
Time until begin control room emergency HVAC	1 minute
Normal ventilation flow rate	
Unfiltered makeup	2750 cfm
Emergency ventilation system flow rates	
Filtered makeup	0 cfm
Filtered recirculation	2500 cfm (+10%)
Unfiltered inleakage	1500 cfm
Control room filter efficiencies	
Elemental	90%
Organic	90%
Particulate	99%
Atmospheric dispersion factors	Table 9

Table 9
Atmospheric Dispersion Factors

Exclusion Area Boundary

Time (hr)	X/Q (sec/m^3)
0 - 2	2.232E-04

Low Population Zone

Time (hr)	X/Q (sec/m^3)
0 - 2	3.977E-05
2 - 24	4.100E-06
24 - 48	2.427E-06
48 - 720	4.473E-07

Control Room

Time (hr)	X/Q (sec/m^3)
0 - 8	2.93E-03
8 - 24	1.73E-03
24 - 48	6.74E-04
48 - 720	1.93E-04