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

Dominion®

January 30, 2006

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

Serial No. 05-750
KPS/LIC/GR: R2
Docket No. 50-305
License No. DPR-43

DOMINION ENERGY KEWAUNEE, INC.
KEWAUNEE POWER STATION
LICENSE AMENDMENT REQUEST 211, "RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE"

Pursuant to 10 CFR 50.90, Dominion Energy Kewaunee, Inc. (DEK) requests approval of a proposed license amendment for the Kewaunee Power Station (Kewaunee). This license amendment request (LAR) would modify the currently approved radiological accident analyses (RAA) and associated technical specifications (TS). The current Kewaunee RAA were approved by letter dated March 17, 2003 (Adams Accession No ML030210062).

This LAR incorporates changes necessary to account for the difference between the control room emergency zone (CREZ) unfiltered in-leakage (UFI) assumed in the current RAA and the CREZ UFI that was measured during testing. In December of 2004, Kewaunee's CREZ was tested to confirm the unfiltered in-leakage assumption used in the approved RAA. The test results showed that the CREZ UFI was greater than that assumed in the approved RAA. As an interim measure, administrative restrictions were placed on other RAA input assumptions to ensure the CREZ remained operable. The resolution to this non-conforming condition is to permanently incorporate the increase in assumed CREZ UFI into the RAA. The increase in the assumed control room unfiltered in-leakage was determined to be a facility change which caused an increase in the dose consequences of the approved RAA. Therefore, this LAR is submitted for approval as required by 10 CFR 50.59 (c)(2).

To support the proposed changes to the RAA input assumptions, DEK also requests approval of changes to several TS. These changes are necessary to implement the proposed change to the RAA. Specifically, changes are requested to TS Section 3.1, "Reactor Coolant System," TS Section 3.6, "Containment System," and TS Section 6.20, "Containment Leakage Rate Testing Program." Appropriate TS Bases changes are also included to address the proposed TS changes.

Attachment 1 to this letter contains a description, a safety evaluation, a significant hazards determination and environmental considerations for the proposed changes. Attachments 2 and 3 contain the marked up and affected TS pages as revised, respectively. Attachments 4 and 5 contain the marked up and affected TS Bases pages, respectively. Attachment 6 contains the proposed revision to Kewaunee accident analyses.

DEK requests approval of this LAR by January 31, 2007. Once approved, this amendment will be implemented within 60 days.

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If you have any questions or require additional information, please contact Mr. Gerald Riste at (920) 388-8424. A complete copy of this submittal has been transmitted to the State of Wisconsin as required by 10 CFR 50.91(b)(1).

Very truly yours,

Leslie N. Hartz
Vice President-Nuclear Engineering

Attachments

1. Evaluation of License Amendment Request 211
2. Marked-up TS Pages For License Amendment Request 211
3. Affected TS Pages For License Amendment Request 211
4. Marked-up TS Bases Pages For License Amendment Request 211
5. Affected TS Bases Pages For License Amendment Request 211
6. Report of Kewaunee Radiological Accident Analyses For Increased Control Room Unfiltered Inleakage

Commitments made by this letter: NONE

cc: Regional Administrator, Region III
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COMMONWEALTH OF VIRGINIA)
)
COUNTY OF HENRICO)

The foregoing document was acknowledged before me, in and for the County and Commonwealth aforesaid, today by Leslie N. Hartz, who is Vice President – Nuclear Engineering of Dominion Energy Kewaunee, Inc. She has affirmed before me that she is duly authorized to execute and file the foregoing document in behalf of that Company, and that the statements in the document are true to the best of her knowledge and belief.

Acknowledged before me this 30th day of January, 2006.

My Commission Expires: My Commission Expires July 31, 2007.

Vera T. Thomas

Notary Public

(SEAL)

ATTACHMENT 1

LICENSE AMENDMENT REQUEST - 211

RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

EVALUATION OF LICENSE AMENDMENT REQUEST - 211

KEWAUNEE POWER STATION

DOMINION ENERGY KEWAUNEE, INC.

EVALUATION OF LICENSE AMENDMENT REQUEST - 211

1.0 DESCRIPTION

Dominion Energy Kewaunee, Inc. (DEK) proposes to amend the Kewaunee Power Station (Kewaunee) radiological accident analyses (RAA) and the operating license DPR-43, Appendix A, "Technical Specifications." This license amendment request (LAR) is being submitted to reconcile assumptions used in Kewaunee's RAA with the results observed in the control room emergency zone (CREZ) unfiltered in-leakage (UFI) tracer gas test performed in response to Generic Letter 2003-01. Specifically, DEK requests approval of the revised RAA based on the results of revised analysis identifying more than a minimal increase in the consequences of an accident previously evaluated. Currently, the Kewaunee CREZ is operable-but-non-conforming based on the measured CREZ boundary UFI being higher than the licensing basis accident analysis assumed CREZ UFI.

2.0 PROPOSED CHANGE

The following changes are proposed in this LAR. Marked up and affected pages showing the proposed changes are located in Attachments 2 and 3, respectively. The corresponding Technical Specifications (TS) bases will be updated appropriately to reflect the changes listed below:

1. Change TS 3.1.c.2.A from 60 μCi/gram DOSE EQUIVALENT I-131 to 20 μCi/gram DOSE EQUIVALENT I-131.

2. Change TS 3.6.c.3.B from $\geq$ 95% radioactive methyl iodide removal to $\geq$ 97.5% radioactive methyl iodide removal.

3. Change TS 6.20 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.

3.0 BACKGROUND

On March 19, 2002, Kewaunee staff submitted a request to revise the Kewaunee design basis radiological accident analysis (ADAMS Accession NO. ML020870565). In this request the control room unfiltered in-leakage was stated as 200 cfm. Subsequently, the NRC staff requested additional information concerning the use of the 200 cfm in a letter dated July 3, 2002. Refer to question number two (ADAMS Accession NO. ML021790648). Kewaunee staff replied on September 13, 2002, (ADAMS Accession NO. ML022680167) stating the basis for the 200 cfm unfiltered in-leakage.

On March 17, 2003, the NRC issued license amendment 166, which approved the implementation of the alternate source term methodology for Kewaunee (ADAMS Accession NO. ML030210062). The safety evaluation associated with license amendment 166 stated that the use of 200 cfm unfiltered in-leakage for the control room was acceptable pending resolution of the control room in-leakage generic issue. Resolution of the generic control room issue

(Generic Letter 2003-01) required a tracer gas test to confirm the assumed control room unfiltered in-leakage. This letter completes the commitment made in Kewaunee's GL 2003-01 supplemental response dated April 1, 2005 (ADAM Accession NO. ML050970303).

Because the measured CREZ UFI was higher than assumed in the radiological accident analyses, the radiological accidents were re-analyzed assuming an increased CREZ UFI that bounds the measured UFI value. The revised radiological accident analyses presented in Attachment 6 support the revised TS proposed by this LAR. The revised analyses results demonstrate that dose consequences are within the acceptable limits described in 10 CFR50.67 and Regulatory Guide 1.183, July 2000.

Proposed Change to TS 3.1.c.2.A, TS 3.6.c.3.B, and TS 6.20

The CREZ boundary is in an operable but non-conforming condition with compensatory measures implemented. The proposed TS changes along with the re-analysis of the radiological accidents provide the bases to remove the operable but non-conforming designation of the Kewaunee CREZ boundary and return it to a fully conforming designation.

4.0 TECHNICAL ANALYSIS

Proposed Change to TS 3.1.c.2.A

Currently, the TS limit for the Reactor Coolant System activity is 60 μCi/gram DOSE EQUIVALENT I-131. Historical chemistry data show that, although more limiting, the proposed conservative change to 20 μCi/gram is reasonable and continues to provide adequate operating margin.

Proposed Change to TS 3.6.c.3.B

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%. Historical test data show that, although more limiting, the proposed conservative change to ≥ 97.5% is reasonable and continues to provide adequate operating margin.

Proposed Change to TS 6.20

The current maximum allowable containment leakage rate is 0.5 weight percent/day of the contained air per 24 hours at a peak test pressure of 46 psig. Historical test data show that, although more limiting, the proposed conservative change to 0.2 weight percent/day is reasonable and continues to provide adequate margin to actual measurements of containment leakage rates.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Dominion Energy Kewaunee, Inc. (DEK) proposes to amend the Kewaunee operating license DPR-43, Appendix A, "Technical Specifications." This license amendment request (LAR) is being submitted to reconcile assumptions used in the Kewaunee Power Station (Kewaunee) radiological accident analysis (RAA) with the results observed in the control room emergency zone (CREZ) unfiltered in-leakage (UFI) tracer gas test performed in response to Generic Letter 2003-01. Currently, the Kewaunee CREZ is operable-but-non-conforming based on the measured CREZ boundary UFI being higher than the licensing basis accident analysis assumed CREZ UFI.

DEK has evaluated whether or not a significant hazards consideration is involved with these proposed amendments 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?**

   No. There are no system, structural, or component (SSC) alterations due to these changes. The radiological accident analyses inputs modified by this request are not accident initiators and do not affect the frequency of occurrence of previously analyzed transients.

   The radiological accident analyses have demonstrated acceptable results using the revised inputs for all affected accidents. Further, the proposed changes do not alter or prevent the ability of structures, systems or components to perform their intended function to mitigate the consequences of accidents previously evaluated in the Updated Safety Analysis Report.

   Therefore, the changes do 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?**

   No. There are no physical changes to the plant SSCs and there is no adverse impact on component or system interactions due to the proposed changes. The modes of operation of the plant remain unchanged and the design functions of all the safety systems remain in compliance with the applicable safety analysis acceptance criteria. Therefore, the proposed changes do 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?**

No. The radiological accident analysis inputs modified by this request were incorporated into the revised radiological accident analyses. The revised radiological analyses satisfy all applicable acceptance criteria. There is no adverse effect on plant safety due to this proposed license amendment. Therefore, the change does not involve a significant reduction in the margin of safety.

Conclusion

Therefore, it is concluded the proposed changes present 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 Kewaunee Power Station was designed, constructed, and is operated in accordance with the intent of the Atomic Energy Commission (AEC) General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the construction of the plant was about 50% completed prior to the issuance of the February 20, 1971, 10 CFR 50 Appendix A General Design Criteria, the plant was not required to be reanalyzed and the Final Safety Analysis Report (FSAR) was not required to be revised to reflect these later criteria. However, the AEC Safety Evaluation Report (SER), issued July 24, 1972, acknowledged that the AEC staff assessed the plant, as described in the FSAR (Amendment No.7), against the Appendix A design criteria and"... are satisfied that the plant design generally conforms to the intent of these criteria."

In a letter dated October 2, 1967, the Atomic Industrial Forum (AIF) distributed comments on the July 1967 AEC GDC. This document proposed changes to the AEC GDC in both wording and content, including deletion of some criteria. The Kewaunee FSAR adopted the AIF document as the Kewaunee Staff's understanding of the method for complying with the AEC GDC, as evidenced by the various sections in the FSAR, which quote the AIF criterion rather than the AEC GDC. However, the AEC SER specifically states that the NRC staff reviewed the plant design against the 1971 version of the AEC GDC, and found the design acceptable for issuing a plant operating license.

In the following discussion, the GDC applicable to Kewaunee are listed with the associated 10 CFR 50 Appendix A GDC criteria listed in parenthesis.

Kewaunee GDC 11 - Control Room (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 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.

Sufficient design features (shielding, distances, and containment integrity and filtration systems) are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, and ingress to and egress from the control room, that exceed established dose acceptance criteria. All of the design basis radiological accidents were addressed in the license amendment request (LAR) for implementing alternate source term (AST) for Kewaunee. In the AST LAR, a revision to the radiological consequence analyses was submitted for the Kewaunee design-basis radiological accident analysis to implement the AST as described in Regulatory Guide 1.183 and pursuant to 10 CFR 50.67, "Accident Source Term."

In support of this request, the dose consequence analyses were revised to show that the control room dose remains below the limits prescribed in 10 CFR 50.67. Subsequently the NRC found the analyses acceptable and issued a safety evaluation approving the use of the AST methodology for Kewaunee. The analyses submitted, using the AST methodology, showed the maximum 30-day whole body dose consequence satisfied the dose acceptance criterion of 5.0 Rem TEDE; therefore, the analyses for Kewaunee were acceptable.

The analyses performed for the stretch power uprate followed the methodology from the AST licensing submittal. The radiological accident analyses performed for the Kewaunee stretch power uprate also satisfied the control room dose acceptance criterion of 5.0 REM and were approved as part of the stretch power uprate license amendment.

In response to Item III.D.3.4 of NUREG-0737, a review of post-accident control room habitability was performed. The NRC determined that the control room habitability systems are acceptable and will provide a safe, habitable environment within the control room under design basis accident radiation and toxic gas conditions, including loss of coolant accidents.

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 the NRC's safety evaluation (SE) associated with this item, the NRC stated that they reviewed the submittals for Kewaunee 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 Regulatory Guides 1.78 and 1.95. The NRC concluded that the design met the criteria identified in NUREG 0737 and was acceptable.

The Kewaunee staff took four exceptions to SRP acceptance criteria. Those 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, and;
4. Requirement to perform a toxic gas, ammonia spill, analysis to determine the affects on control room habitability.

The NRC staff accepted the Kewaunee position for exceptions 1 and 2; concluding that it is sufficient that food supplies and potassium iodide tablets are readily available from nearby sources. The NRC accepted the Kewaunee position on exception 3 based on the condition that for all releases of radioactivity, 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 monitor in the air intake would never be the sole means of isolating that system. Exception 4 was resolved by reporting a re-appraisal of the protection of the control room (CR) against toxic gases. These positions are still in effect.

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 (CRPAR) system. Kewaunee control room isolation and start of CRPAR also occurs on Safety Injection and Steam Exclusion signals. 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.

Kewaunee is considered to be in full compliance with the Kewaunee GDC - 11 as described above. However, the unfiltered in-leakage measured during the subsequent tracer gas testing of the control room envelope exceeded the radiological accident analyses assumptions. As a consequence, the Kewaunee radiological accidents were re-analyzed assuming higher unfiltered in-leakage.

6.0 ENVIRONMENTAL CONSIDERATIONS

This proposed amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or a change to a surveillance requirement. DEK has determined that the proposed amendment involves no significant hazards considerations and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in the individual or cumulative occupational radiation exposure. Accordingly, this proposed 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 this proposed amendment.

7.0 REFERENCES

7.1. Letter from John G. Lamb (NRC) to Thomas Coutu, "Kewaunee Nuclear Power Plant - Issuance Of Amendment Regarding Implementation Of Alternate Source Term (TAC NO. MB4596)," dated March 17, 2003. (Adams Accession NO. ML030210062)

7.2. Letter from Craig W. Lambert to Document Control Desk, "Generic Letter 2003-01: Control Room Habitability - Supplemental Response," dated April 1, 2005. (ADAMS Accession NO. ML050970303)

7.3. Letter from John G. Lamb (NRC) to Thomas Coutu, "Kewaunee Nuclear Power Plant - Issuance of Amendment Regarding Stretch Power Uprate (TAC NO. MB9031)," dated February 27, 2004. (ADAMS Accession NO. ML04030633)

7.4. Letter from Thomas Coutu to Document Control Desk, "License Amendment Request 195, Application for Stretch Power Uprate For Kewaunee Nuclear Power Plant," dated May, 22, 2003. (ADAMS Accession NO. ML031500724)

ATTACHMENT 2

LICENSE AMENDMENT REQUEST 211

RADIOLOGICAL ACCIDENT ANALYSIS AND
ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

MARKED UP TS PAGES FOR LICENSE AMENDMENT REQUEST 211

KEWAUNEE POWER STATION

MARKED UP TS PAGES:

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

DOMINION ENERGY KEWAUNEE, INC.

B. A vent pathway shall be provided with an effective flow cross section ≥ 6.4 square inches.

1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

A. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

A. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding ~~60~~ 20 μCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

C. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 *or* $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

3. Performance Requirements

A. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

B. The results of laboratory carbon sample analysis from the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System carbon shall show ≥ ~~95~~97.5% radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C, 95% RH for the Shield Building Ventilation System and 30°C, 95% RH for the Auxiliary Building Special Ventilation System.

C. Fans shall operate within ± 10% of design flow when tested.

d. If the internal pressure of the reactor containment vessel exceeds 2 psi, the condition shall be corrected within 8 hours or the reactor shall be placed in a subcritical condition.

e. The reactor shall not be taken above the COLD SHUTDOWN condition unless the containment ambient temperature is > 40°F.

6.20 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The provisions of TS 4.0.b do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The provisions of TS 4.0.c are applicable to the Containment Leakage Rate Testing Program.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident is less than the containment internal test pressure, P_a. The maximum allowable leakage rate (L_a) is 0.~~5~~2 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig.

For penetrations which extend into the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.10L_a$. For penetrations which are exterior to both the shield building and the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.01L_a$. If leak rates are exceeded, repairs and retest shall be performed to demonstrate reduction of the combined leak rate to these values.

Leakage rate acceptance criteria:

a. The containment leakage rate acceptance criterion is $\leq 1.0L_a$.

b. Prior to unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.6L_a$ for Type B and C tests and $< 0.75L_a$ for the Type A test.

c. The personnel and emergency air lock leakage rates, when combined with the cumulative Type B and C leakage, shall be $< 0.6L_a$. For each air lock door seal, the leakage rate shall be $< 0.005L_a$ when tested to ≥ 10 psig.

ATTACHMENT 3

LICENSE AMENDMENT REQUEST 211

RADIOLOGICAL ACCIDENT ANALYSIS AND
ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

AFFECTED TS PAGES FOR LICENSE AMENDMENT REQUEST 211

KEWAUNEE POWER STATION

AFFECTED TS PAGES:

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

DOMINION ENERGY KEWAUNEE, INC.

B. A vent pathway shall be provided with an effective flow cross section ≥ 6.4 square inches.

   1. When low temperature overpressure protection is provided via a vent pathway, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position. If the vent path is provided by any other means, then verify the vent pathway every 12 hours.

c. Maximum Coolant Activity

1. The specific activity of the reactor coolant shall be limited to:

   A. ≤ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\overline{E}$ is the average sum of the beta and gamma energies in Mev per disintegration) whenever the reactor is critical or the average coolant temperature is > 500°F.

2. If the reactor is critical or the average temperature is > 500°F:

   A. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding 20 μCi/gram DOSE EQUIVALENT I-131, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within six hours.

   B. With the specific activity of the reactor coolant $> \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within six hours.

   C. With the specific activity of the reactor coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 $or > \frac{91}{\overline{E}} \frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every four hours until restored to within its limits.

3. Annual reporting requirements are identified in TS 6.9.a.2.D.

3. Performance Requirements

A. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show ≥ 99% DOP removal and ≥ 99% halogenated hydrocarbon removal.

B. The results of laboratory carbon sample analysis from the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System carbon shall show ≥ 97.5% radioactive methyl iodide removal when tested in accordance with ASTM D3803-89 at conditions of 30°C, 95% RH for the Shield Building Ventilation System and 30°C, 95% RH for the Auxiliary Building Special Ventilation System.

C. Fans shall operate within ± 10% of design flow when tested.

d. If the internal pressure of the reactor containment vessel exceeds 2 psi, the condition shall be corrected within 8 hours or the reactor shall be placed in a subcritical condition.

e. The reactor shall not be taken above the COLD SHUTDOWN condition unless the containment ambient temperature is > 40°F.

6.20 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. The provisions of TS 4.0.b do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program. The provisions of TS 4.0.c are applicable to the Containment Leakage Rate Testing Program.

The peak calculated containment internal pressure for the design basis loss-of-coolant accident is less than the containment internal test pressure, P_a. The maximum allowable leakage rate (L_a) is 0.2 weight percent of the contained air per 24 hours at the peak test pressure (P_a) of 46 psig.

For penetrations which extend into the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.10L_a$. For penetrations which are exterior to both the shield building and the auxiliary building special ventilation zone, the combined leak rate from these penetrations shall not exceed $0.01L_a$. If leak rates are exceeded, repairs and retest shall be performed to demonstrate reduction of the combined leak rate to these values.

Leakage rate acceptance criteria:

a. The containment leakage rate acceptance criterion is $\leq 1.0L_a$.

b. Prior to unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.6L_a$ for Type B and C tests and $< 0.75L_a$ for the Type A test.

c. The personnel and emergency air lock leakage rates, when combined with the cumulative Type B and C leakage, shall be $< 0.6L_a$. For each air lock door seal, the leakage rate shall be $< 0.005L_a$ when tested to ≥ 10 psig.

ATTACHMENT 4

LICENSE AMENDMENT REQUEST 211

RADIOLOGICAL ACCIDENT ANALYSIS AND
ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

MARKED UP TS PAGES FOR LICENSE AMENDMENT REQUEST 211

KEWAUNEE POWER STATION

MARKED UP TS BASES PAGES:

TS B3.1-8
TS B3.6-4

DOMINION ENERGY KEWAUNEE, INC.

Maximum Coolant Activity (TS 3.1.c)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is limited to $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[15] are analyzed assuming an RCS activity of 1.0 μCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of ~~60~~ 20 μCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 50.67 limits.

The action of reducing average reactor coolant temperature to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

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(15) USAR Section 14.0

Ventilation Systems (TS 3.6.c)

Proper functioning of the Shield Building Ventilation System is essential to the performance of the Containment System. Therefore, except for reasonable periods of maintenance outage for one redundant train of equipment, the complete system should be in readiness whenever CONTAINMENT SYSTEM INTEGRITY is required. Proper functioning of the Auxiliary Building Special Ventilation System is similarly necessary to preclude possible unfiltered leakage through penetrations that enter the Special Ventilation Zone (Zone SV).

Both the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System are designed to automatically start following a safety injection signal. Each of the two trains of both systems has 100% capacity. If one train of either system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue while repairs are being made. If both trains of either system are inoperable, the plant will be brought to a condition where the air purification system would not be required.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine release to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodine removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. The performance criteria for the safeguard ventilation fans are stated in Section 5.5 and 9.6 of the USAR. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

Accident analysis assumes a charcoal adsorber efficiency of ~~90~~95%.[2] To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of ~~90~~95%, this equates to a methyl iodide penetration of ~~10~~5%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to ~~5~~2.5%. Thus, the acceptance criteria of ~~95~~97.5% efficient will be used for the charcoal adsorbers.

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[2] USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

ATTACHMENT 5

LICENSE AMENDMENT REQUEST 211

RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

AFFECTED TS BASES PAGES FOR LICENSE AMENDMENT REQUEST 211

KEWAUNEE POWER STATION

AFFECTED TS BASES PAGES:

TS B3.1-8
TS B3.6-4

DOMINION ENERGY KEWAUNEE, INC.

Maximum Coolant Activity (TS 3.1.c)

The maximum dose that an individual may receive following an accident is specified in GDC 19 and 10 CFR 50.67. The limits on maximum coolant activity ensure that the calculated doses are held to the limits specified in GDC 19 and to a fraction of the 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is limited to $\leq$ 1.0 µCi/gram DOSE EQUIVALENT I-131 to ensure the dose does not exceed the GDC-19 and 10 CFR 50.67 guidelines. The applicable accidents identified in the USAR[15] are analyzed assuming an RCS activity of 1.0 µCi/gram DOSE EQUIVALENT I-131 incorporating an accident initiated iodine spike when required. To ensure the conditions allowed are taken into account, the applicable accidents are also analyzed considering a pre-existing iodine spike of 20 µCi/gram DOSE EQUIVALENT I-131. The results obtained from these analyses indicate that the control room and off-site doses are within the acceptance criteria of GDC-19 and a fraction of 10 CFR 50.67 limits.

The Reactor Coolant Specific Activity is also limited to a gross activity of $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. Again the accidents under consideration are analyzed assuming a gross activity of $\frac{91}{\bar{E}} \frac{\mu Ci}{cc}$. The results obtained from these analyses indicate the control room and off-site dose are within the acceptance criteria of GDC-19 and a small fraction of 10 CFR 50.67 limits.

The action of reducing average reactor coolant temperature to < 500°F prevents the release of activity should a steam generator tube rupture occur since the saturation pressure of the reactor coolant is below the lift pressure of the main steam safety valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

(15) USAR Section 14.0

Ventilation Systems (TS 3.6.c)

Proper functioning of the Shield Building Ventilation System is essential to the performance of the Containment System. Therefore, except for reasonable periods of maintenance outage for one redundant train of equipment, the complete system should be in readiness whenever CONTAINMENT SYSTEM INTEGRITY is required. Proper functioning of the Auxiliary Building Special Ventilation System is similarly necessary to preclude possible unfiltered leakage through penetrations that enter the Special Ventilation Zone (Zone SV).

Both the Shield Building Ventilation System and the Auxiliary Building Special Ventilation System are designed to automatically start following a safety injection signal. Each of the two trains of both systems has 100% capacity. If one train of either system is found to be inoperable, there is not an immediate threat to the containment system performance and reactor operation may continue while repairs are being made. If both trains of either system are inoperable, the plant will be brought to a condition where the air purification system would not be required.

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine release to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon and DOP respectively. The laboratory carbon sample test results indicate a radioactive methyl iodine removal efficiency under test conditions which are more severe than accident conditions.

Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. The performance criteria for the safeguard ventilation fans are stated in Section 5.5 and 9.6 of the USAR. If the performances are as specified, the calculated doses would be less than the guidelines stated in 10 CFR Part 100 for the accidents analyzed.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

Accident analysis assumes a charcoal adsorber efficiency of 95%.[2] To ensure the charcoal adsorbers maintain that efficiency throughout the operating cycle, a safety factor of 2 is used. Therefore, if accident analysis assumes a charcoal adsorber efficiency of 95%, this equates to a methyl iodide penetration of 5%. If a safety factor of 2 is assumed, the methyl iodide penetration is reduced to 2.5%. Thus, the acceptance criteria of 97.5% efficient will be used for the charcoal adsorbers.

[2] USAR TABLE 14.3-8, "Major Assumptions for Design Basis LOCA Analysis"

ATTACHMENT 6

LICENSE AMENDMENT REQUEST 211

RADIOLOGICAL ACCIDENT ANALYSIS AND ASSOCIATED TECHNICAL SPECIFICATIONS CHANGE

REVISED RADIOLOGICAL ACCIDENT ANALYSES

DOMINION ENERGY KEWAUNEE, INC.

Report of Kewaunee Radiological Accident Analyses For Increased Control Room Unfiltered Inleakage

1.0 Introduction and Background

The following radiological accidents are the design basis accidents (DBAs) for the Kewaunee Power Station (KPS):

- Main steam line break (MSLB)
- Locked reactor coolant pump (RCP) rotor
- Rod ejection (RE)
- Steam generator tube rupture (SGTR)
- Large-break loss-of-coolant accident (LBLOCA)
- Waste gas decay tank (GDT) rupture
- Volume control tank (VCT) rupture
- Fuel-handling accident (FHA)

For each accident, the total effective dose equivalent (TEDE) doses are determined at the site boundary (SB) for the limiting 0- to 2-hour period, at the low-population zone (LPZ) boundary for the duration of the accident and in the control room for 30 days. The offsite dose acceptance limits are based on 10CFR50.67 criteria. Depending on the event, the acceptance limit is 100 percent of 10CFR50.67 or a fraction of these acceptance limits. The control room dose acceptance limit from 10CFR50.67 is 5.0 rem TEDE.

Alternate Source Term (AST) analytical methods and assumptions outlined in Regulatory Guide 1.183 (Reference 3) were approved for KPS in license amendment #166 (Reference 1). The radiological accident analyses performed for the KPS stretch power uprate followed the approved methodology from the AST license amendment. The radiological accident analyses for the stretch power uprate were approved as part of the stretch power uprate license amendment # 172 (Reference 2).

Tracer gas in-leakage tests were performed in December 2004 by NUCON International Inc. (Reference 4). The amount of air in-leakage into the Control Room Emergency Zone (CREZ) was evaluated using the concentration decay method under isolation conditions. This test is based on ASTM E 741 and conducted to ensure compliance with the US NRC Generic Letter 2003-01. Kewaunee's official response to Generic Letter 2003-01 (Reference 5) addressed the Generic Letter 2003-01 request: "Perform the ASTM E741 testing and, provide the requested response to Generic Letter Item 1(a)." ASTM E741 baseline testing results were provided to the NRC in Enclosure 1 of Reference 5.

Two concentration decay tests were performed to determine total unfiltered in-leakage, one with Control Room Post Accident Recirculation (CRPAR) Train A operating and one with CRPAR Train

B operating. The following results were obtained for total unfiltered in-leakage to the three rooms contained in the control room envelope.

Control Room Emergency Zone Inleakage Test Results		
Date of Test	Train Tested	**Total Inleakage**
14 December 2004	CRPAR Train A	409 ±29 cfm
15 December 2004	CRPAR Train B	447±51 cfm

The tracer gas in-leakage test showed that the current radiological accident analysis CREZ unfiltered in-leakage assumption of 200 cfm was not met. An operability determination was performed that evaluated CREZ unfiltered in-leakage in excess of that which was assumed in the analysis. The operability determination specified revised administrative limits of containment leak rate, reactor coolant system activity, and carbon filter absorption efficiency affecting radiological source and potential radiological release pathways. These administrative limits compensate for the higher measured control room in-leakage and are currently implemented in the appropriate plant procedures. The administrative limits, developed from radiological accident analysis sensitivity cases, ensure that the radiological dose consequences remain within the limits specified in the current licensing basis acceptance criteria of 10 CFR 50.67, including the limitations of Regulatory Guide 1.183. No credit was taken in the operability determination for the use of self-contained breathing apparatus or potassium iodide.

Because the measured CREZ UFI is higher than assumed in the current licensing basis radiological accident analyses, the radiological accidents were re-analyzed assuming increased CREZ UFI. The CREZ UFI assumption in the analyses is increased to a value that bounds the measured CREZ UFI including uncertainties and provides sufficient operating margin. The re-analysis of the radiological accidents at the higher CREZ UFI is required for full qualification of the KPS CREZ boundary.

The revised radiological accident analyses are consistent with and support the proposed revisions to the technical specifications of this License Amendment Request. The revised analyses require NRC approval since they result in more than a minimal increase in the consequences of an accident previously evaluated in the USAR.

1.1 Input Parameters and Assumptions

The assumptions and inputs used for the revised radiological accident analyses are the same as the assumptions and inputs of the current approved radiological accident analyses (Reference 2) except as described below and as presented for each accident in subsections 2.0 through 9.0.

The CREZ UFI is increased to at least 800 cfm for events that model control room isolation on a safety injection signal (i.e. large break loss of coolant accident (LBLOCA), rod ejection (RE), main steam line break (MSLB) and steam generator tube rupture (SGTR).

For these accidents, the following revised analysis assumptions have been employed in the analyses.

- The control room damper closure time is increased to 20 seconds from the current analysis value of 10 seconds. The increase in damper closure time bounds actual damper measured closure times and provides sufficient operating margin.
- The control room unfiltered inleakage during emergency mode HVAC operation is increased to at least 800 cfm, from the current analysis value of 200 cfm. The CREZ UFI assumption bounds the measured CREZ UFI, including uncertainties, with sufficient operating margin.
- For the LOCA analysis, the shield building and auxiliary building filter efficiency for elemental and organic iodine is increased to 95%, from the current analysis value of 90%. The revised filter efficiency limits are based on radiological accident analysis sensitivity cases. The revised limits bound plant charcoal filter test results and provide sufficient operating margin.
- For the LOCA analysis, the containment leak rate is reduced to 0.2 wt%/day from the current analysis value of 0.5 wt%/day. The reduced containment leak rate limit is based on radiological accident analysis sensitivity cases. The revised containment leak rate limit bounds the plant measured containment leak rate test results and provides sufficient operating margin.
- For the SGTR analysis, the pre-accident iodine spike reactor coolant iodine concentration is reduced from 60 μCi/gm dose equivalent I-131 to 20 μCi/gm dose equivalent I-131. The other events that model the pre-accident iodine spike activity retain the 60 μCi/gm dose equivalent I-131. The reduced pre-accident iodine spike limit is based on radiological accident analysis sensitivity cases. The revised limit bounds plant reactor coolant system activity measurements and provides sufficient operating margin.

The CREZ UFI is increased to at least 1500 cfm for events that model control room isolation on a control room radiation monitor (RM) R-23 high control room duct activity monitor actuation (i.e., locked rotor (LR) and fuel handling accident (FHA)). This increased inleakage addresses control room isolation damper positioning based on the R-23 monitor. The CR radiation monitor R-23 does not close all CR isolation dampers. Control room isolation dampers 10, 11, 20, and 21 remain open following R-23 high radiation signal. Unfiltered inleakage is increased in the RM R-23 activated cases to compensate for these four dampers remaining open. The gas decay tank and volume control tank rupture analyses have been shown to be bounding because minimizing CREZ UFI maximizes the calculated control room dose (Reference 2) and are therefore not included in this re-analysis effort.

The following revised analysis assumptions have been employed in the analyses. These assumptions are conservative and bound actual plant performance data.

- The control room damper closure time is increased to 20 seconds, from the current analysis value of 10 seconds. The increase in damper closure time bounds actual damper measured closure times and provides sufficient operating margin.
- A range of control room unfiltered inflow during normal mode HVAC operation from 1620 cfm to 2750 cfm is considered to determine the impact of the flow rate on the R-23 isolation signal timing and the calculated doses. The range of control room unfiltered flow bounds actual measured CR HVAC system flows and provides sufficient operating margin.
- The control room unfiltered inleakage during emergency mode HVAC operation is increased to 1500 cfm, from the current analysis value of 200 cfm. The CREZ UFI assumption of 1500 cfm bounds the measured CREZ UFI and the additional UFI attributable to CR dampers 10, 11, 20, and 21 remaining open following CR isolation on RM R-23, including uncertainties, with sufficient operating margin.

- The locked rotor fuel failure assumption is revised. Rather than assuming 100% of the fuel rods fail and release their gap activity, the analysis will assume that 50% of the fuel rods fail and release their gap activity based on the Reload Safety Analysis limit for rods-in departure from nucleate boiling (DNB). The percent of rods-in DNB is a Reload Safety Analysis limit with a current limit value of 50%. This reload safety analysis limit is confirmed for each reload. The reload analysis conservatively applies the maximum peaking factor of 1.7 to all rods assumed to fail to determine gap activity.
- The locked rotor analysis is revised to credit an increase in secondary side fluid mass following reactor trip and auxiliary feedwater initiation. The minimum water mass required to ensure that the steam generator tubes are covered (i.e., steam generator narrow range level greater than zero percent narrow range span) is credited to dilute the secondary activity starting 30 minutes into the event. Steam generator level is in the narrow range post-accident based on operator instructions in the KPS emergency operating procedures.

A summary of the changes to the assumptions and input parameters associated with the control room model used in the analyses is provided in Table 1.

The current baseline radiological accident analyses are bounding with respect to offsite doses. However, offsite doses will be reported for the radiological accident re-analyses.

2.0 Main Steam Line Break Accident

The complete severance of a main steam line outside containment is assumed to occur. The affected steam generator will rapidly depressurize and release iodine activity initially contained in the secondary coolant and primary coolant activity (iodine's and noble gases) transferred via steam generator tube leaks, directly to the outside atmosphere.

A portion of the iodine activity initially contained in the intact steam generator and the activity transferred to the secondary coolant due to tube leakage is released to the atmosphere through either the atmospheric relief valves (ARVs) or the safety valves. The steam line break outside containment will bound any break inside containment since the outside containment break provides a means for direct release to the environment. This section describes the assumptions and analyses performed to determine the offsite and control room doses resulting from the release of activity associated with this event.

2.1 Input Parameters and Assumptions

A summary of the changes to the assumptions and input parameters used in the analysis are itemized in Tables 1 and 2.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is not impacted by the assumed normal mode unfiltered makeup flow. The Reference 2 assumption of control room isolation at 5 minutes remains conservative with the increased damper closure time of 20 seconds.

2.2 Acceptance Criteria

The offsite dose limit for MSLB with a pre-accident iodine spike is 25 rem TEDE per Regulatory Guide 1.183. This is the guideline value of 10CFR50.67. For MSLB with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per Regulatory Guide 1.183. This is 10 percent of the guideline value of 10CFR50.67. The limit for control room dose is 5.0 rem TEDE for both cases per 10CFR50.67.

2.3 Results and Conclusions

The doses due to the MSLB with a pre-accident iodine spike are:

MSLB PRE-ACCIDENT IODINE SPIKE CASE	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.03	0.03	25
LPZ	0.01	0.01	25
Control Room	0.70	0.50	5

The doses due to the MSLB with an accident-initiated iodine spike are:

MSLB ACCIDENT-INITIATED IODINE SPIKE CASE	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.06	0.06	2.5
LPZ	0.02	0.02	2.5
Control Room	2.60	1.00	5

The SB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours for the pre-accident iodine spike, and from 4 to 6 hours for the accident-initiated iodine spike.

The acceptance criteria are met.

3.0 Locked-Rotor Accident

An instantaneous seizure of a reactor coolant pump (RCP) rotor is assumed to occur, which rapidly reduces flow through the affected reactor coolant loop (RCL). Fuel-cladding damage may be predicted as a result of this accident. Due to the pressure differential between the primary and

secondary systems and assumed steam generator tube leakage, fission products transfer from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the atmospheric relief valves, or safety valves. In addition, iodine activity is contained in the secondary coolant prior to the accident, and some of this activity is released to the atmosphere as a result of steaming from the steam generators following the accident.

3.1 Input Parameters and Assumptions

A summary of the changes to the assumptions and input parameters used in the analysis are itemized in Tables 1 and 3.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is potentially impacted by the assumed normal mode unfiltered makeup flow. The analysis conservatively delays the time of control room isolation by 45 minutes. This isolation timing assumption conservatively bounds the time for generation of the R-23 radiation monitor signal and CR isolation assuming the minimum normal mode unfiltered makeup flow. Analysis also shows that control room area radiation monitor R-1 will alarm early in the accident alerting the operators to a radiological event. Other radiation monitors (e.g. R-9 in the reactor coolant letdown system) will also alarm in the control room due to the increased radioactivity in the RCS.

3.2 Acceptance Criteria

The offsite dose limit for a locked rotor is 2.5 rem TEDE per Regulatory Guide 1.183. This is 10 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE, per 10CFR50.67.

3.3 Results and Conclusions

The doses due to the locked rotor, including the multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event are:

LOCKED ROTOR	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.40	0.50	2.5
LPZ	0.06	0.08	2.5
Control Room	3.90	1.40	5

The SB dose reported is for the worst 2-hour period, determined to be from 6 to 8 hours.

The acceptance criteria are met.

4.0 Rod-Ejection Accident

It is assumed that a control rod drive mechanism (CRDM) pressure housing mechanical failure has occurred, resulting in the ejection of a rod cluster control assembly and drive shaft. As a result of the accident, some fuel cladding damage and a small amount of fuel melting (pellet centerline) are assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive primary coolant is assumed to leak from the primary into the secondary system. A portion of this radioactivity is released to the outside atmosphere through the main condenser, the atmospheric relief valves, or the safety valves. Also, some iodine and alkali metal group activity that is contained in the secondary coolant prior to the accident is released to the atmosphere as a result of steaming from the steam generators following the accident. Finally, radioactive primary coolant is discharged to the containment from the opening in the reactor vessel head. A portion of this radioactivity is released through containment leakage to the environment.

4.1 Input Parameters and Assumptions

A summary of changes to the input parameters and assumptions is provided in Tables 1 and 4.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is not impacted by the assumed normal mode unfiltered makeup flow. The Reference 2 assumption of control room isolation at 2.5 minutes remains conservative with the increased damper closure time of 20 seconds.

4.2 Acceptance Criteria

The offsite dose limit for a rod ejection is 6.3 rem TEDE per Regulatory Guide 1.183. This is approximately 25 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

4.3 Results and Conclusions

The offsite doses due to the rod-ejection accident, including the multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event are:

ROD EJECTION	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.40	0.40	6.3
LPZ	0.09	0.09	6.3
Control Room	4.54	1.91	5

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

5.0 <u>Steam Generator Tube Rupture Translent</u>

The accident analyzed is the double-ended rupture of a single steam generator tube.

It is assumed that the primary-to-secondary break flow following an SGTR results in depressurization of the Reactor Coolant System (RCS), and that reactor trip and safety injection (SI) are automatically initiated on low pressurizer pressure. Loss-of-offsite power (LOOP) is assumed to occur at reactor trip resulting in the release of steam to the atmosphere via the steam generator atmospheric relief valves (ARVs) and/or safety valves. After plant trip and SI actuation it is assumed that the RCS pressure stabilizes and the break flow equilibrates at the point where incoming safety injection flow is balanced by outgoing break flow.

The radiological consequence associated with a steam generator tube rupture (SGTR) event is the transfer of radioactive reactor coolant to the secondary side of the ruptured steam generator and subsequent release of radioactivity to the atmosphere. The primary thermal-hydraulic parameters which affect the calculation of doses for an SGTR include the amount of reactor coolant transferred to the secondary side of the ruptured steam generator, the amount of primary to secondary break flow that flashes to steam and the amount of steam released from the ruptured steam generator to the atmosphere.

5.1 Input Parameters and Assumptions

A summary of the changes to the assumptions and input parameters used in the analysis are itemized in Tables 1 and 5.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is not impacted by the assumed normal mode unfiltered makeup flow. The Reference 2 assumption of control room isolation at 5 minutes remains conservative with the increased damper closure time of 20 seconds.

5.2 Acceptance Criteria

The offsite dose limit for a SGTR with a pre-accident iodine spike is 25 rem TEDE per Regulatory Guide 1.183. This is the guideline value of 10CFR50.67. For a SGTR with an accident-initiated iodine spike, the offsite dose limit is 2.5 rem TEDE per Regulatory Guide 1.183. This is 10 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

5.3 Results and Conclusions

The doses due to the SGTR with a pre-accident iodine spike are:

SGTR PRE-ACCIDENT IODINE SPIKE CASE	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.50	1.30	25
LPZ	0.10	0.30	25
Control Room	1.90	3.10	5

The doses due to the SGTR with an accident-initiated iodine spike are:

SGTR ACCIDENT-INITIATED IODINE SPIKE CASE	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.80	0.80	2.5
LPZ	0.20	0.20	2.5
Control Room	2.80	1.00	5

The SB doses reported are for the worst 2-hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

6.0 Large-Break Loss-of-Coolant Accident

A loss-of-coolant accident (LOCA) is defined as a rupture of the Reactor Coolant System (RCS) piping or of any line connected to the RCS. For the purposes of assessing LOCA post-accident dose, an extremely improbable double-ended rupture of a 29-inch inside diameter pipe in the reactor coolant loop is assumed. In accordance with Regulatory Guide 1.183, it is assumed that various portions of core activity are released to the Reactor Containment Vessel. Releases of this magnitude cannot occur unless there are multiple failures, which is beyond the typical design basis accident (DBA) that considers a single active failure. Activity from the core is assumed to be released to the containment and then to the environment by containment leakage and leakage from the Emergency Core Cooling System (ECCS) as it re-circulates sump solution outside the containment. Details regarding the Loss-of-Coolant dose analysis assumptions may be found in Kewaunee FSAR section 14.3.5.

6.1 Input Parameters and Assumptions

A summary of the changes to the assumptions and input parameters used in the analysis are itemized in Tables 1 and 6.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is not impacted by the assumed normal mode unfiltered makeup flow. The Reference 2 assumption of control room isolation at 2 minutes remains conservative with the increased damper closure time of 20 seconds.

6.2 Acceptance Criteria

The offsite dose limit for a LOCA is 25 rem TEDE per Regulatory Guide 1.183. This is the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

6.3 Results and Conclusions

The calculated offsite and control room doses, including the multiplier developed to bound variations in core average enrichment, core mass, and cycle length for this event and including the direct and sky-shine dose for the 30-day duration are:

LOCA	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.52	1.31	25
LPZ	0.09	0.22	25
Control Room	4.95	4.58	5

The SB dose reported is for the worst 2-hour period, determined to be from 1.8 to 3.8 hours.

The acceptance criteria are met.

7.0 Gas Decay Tank Rupture

The GDT rupture analysis has been shown to be bounding assuming minimum CREZ UFI (Reference 2, see Section 3.5.2.8 of the attached SER). Because of the short duration of the release, minimizing the assumed CREZ UFI maximizes the calculated control room dose for this event. This is due to less dilution of the radioactivity in the control room. Therefore, re-analysis at a higher CREZ UFI is not necessary because the current analysis remains bounding.

8.0 Volume Control Tank Rupture

The VCT rupture analysis has been shown to be bounding assuming minimum CREZ UFI (Reference 2, see Section 3.5.2.8 of the attached SER). Because of the short duration of the release, minimizing the assumed CREZ UFI maximizes the calculated control room dose for this event. This is due to less dilution of the radioactivity in the control room. Therefore, re-analysis at the higher CREZ UFI is not necessary because the current analysis remains bounding.

9.0 Fuel-Handling Accident

A fuel assembly is assumed to be dropped and damaged during refueling. Analysis of the accident is performed with assumptions selected so that the results are bounding for the accident occurring either inside containment or in the auxiliary building. Activity released from the damaged assembly is released to the outside atmosphere through either the shield building ventilation system or the Spent Fuel Pool Ventilation System.

9.1 Input Parameters and Assumptions

A summary of the changes to the assumptions and input parameters used in the analysis are itemized in Tables 1 and 7.

The analysis models maximum unfiltered makeup flow during normal ventilation operation to maximize the activity entering the control room, consistent with Reference 2. The control room isolation time is potentially impacted by the assumed normal mode unfiltered makeup flow. The analysis conservatively delays the time of control room isolation by 25 minutes. This isolation timing assumption conservatively bounds the time for generation of the R-23 radiation monitor signal and automatic CR isolation even with the minimum normal mode unfiltered makeup flow. Analysis also shows that control room area radiation monitor R-1 will alarm early in the accident alerting the operators to the radiological event. Other radiation monitors (e.g. containment area

monitor R-2 or auxiliary building spent fuel pool monitor R-5) will also alarm in the control room due to the increased radioactivity in the affected area.

9.2 Acceptance Criteria

The offsite-dose limit for a FHA is 6.3 rem TEDE per Regulatory Guide 1.183. This is approximately 25 percent of the guideline value of 10CFR50.67. The limit for the control room dose is 5.0 rem TEDE per 10CFR50.67.

9.3 Results and Conclusions

The doses due to the FHA are:

FHA	REVISED TEDE DOSE (rem)	CURRENT TEDE DOSE (rem)	ACCEPTANCE CRITERIA (rem TEDE)
SB	0.70	0.70	6.3
LPZ	0.11	0.11	6.3
Control Room	4.90	1.00	5

The SB dose reported is for the worst 2-hour period, determined to be from 0 to 2 hours.

The acceptance criteria are met.

10.0 References

1. 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.

2. 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.

3. Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2000.

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

5. Letter from Craig Lambert (NMC) to USNRC dated April 1, 2005 submitting " Generic Letter 2003-01:Control Room Habitability-Supplemental Response"

Table 1

Revised Control Room Parameters

	REVISED ASSUMPTION	CURRENT ASSUMPTION
Normal Ventilation Flow Rates		
Unfiltered Makeup Flow Rate	1620 - 2750 scfm	2250 - 2750 scfm
Emergency Ventilation Flow Rates		
Unfiltered In-leakage Following SI	≥ 800 scfm	200 scfm
Unfiltered In-leakage Following R-23	1500 scfm	200 scfm
Control Room Isolation Damper Closure Time	20 seconds	10 seconds

Table 2

Revised Assumptions for Main Steam Line Break Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
CREZ Unfiltered In-leakage Flow Rate	1000 scfm	200 scfm

Table 3

Revised Assumptions for Locked Rotor Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
Fraction of Fuel Rods in Core Failing	50%	100%
Peaking Factor Applied to Calculate Activity in Failed Fuel Rods	1.7	Not Applicable
Secondary Coolant Mass 0 to 30 minutes	7.89 E7 gm	7.89 E7 gm
Secondary Coolant Mass >30 minutes	1.06 E8 gm	7.89 E7 gm
CREZ Unfiltered In-leakage Flow Rate	1500 scfm	200 scfm
Time to Isolate Control Room and Start Crediting Emergency Control Room HVAC	45 minutes	10 minutes

Table 4

Revised Assumptions for Rod Ejection Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
CREZ Unfiltered In-leakage Flow Rate	1000 scfm	200 scfm

Table 5

Revised Assumptions for SGTR Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
CREZ Unfiltered In-leakage Flow Rate	1000 scfm	200 scfm
Primary Coolant Iodine Activity Prior to Accident		
Pre-Accident Iodine Spike	20 µCi/gm DE I-131	60 µCi/gm DE I-131
Accident-Initiated Iodine Spike	1 µCi/gm DE I-131	1 µCi/gm DE I-131

Table 6

Revised Assumptions for LBLOCA Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
Shield Building and Auxiliary Building Filter Efficiencies		
Elemental	95%	90%
Organic	95%	90%
Containment Leak Rates		
0 – 24 hours	0.2 (weight %/day)	0.5 (weight %/day)
> 24 hours	0.1 (weight %/day)	0.25 (weight %/day)
CREZ Unfiltered In-leakage Flow Rate	800 scfm	200 scfm

Table 7
Revised Assumptions for FHA Dose Analysis

	REVISED ASSUMPTION	CURRENT ASSUMPTION
CREZ Unfiltered In-leakage Flow Rate	1500 scfm	200 scfm
Time to Isolate Control Room and Time to Start Crediting Emergency Control Room HVAC	25 minutes	1 minute