March 17, 2003

Mr. Thomas Coutu
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 State Highway 42
Kewaunee, WI 54216

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - ISSUANCE OF AMENDMENT REGARDING IMPLEMENTATION OF ALTERNATE SOURCE TERM (TAC NO. MB4596)

Dear Mr. Coutu:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 166 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the current radiological consequence analyses for the KNPP design-basis accidents in response to your application dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002.

The amendment revises the current radiological consequence analyses for the KNPP design-basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" and pursuant to 10 CFR 50.67, "Accident Source Term." You did not request any changes to the current Technical Specifications for KNPP or any modifications to the plant design at this time.

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

Sincerely,

/RA/

John G. Lamb, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures:
1. Amendment No. 166 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

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Distribution w/encls:

PUBLIC	WBeckner, TSB	PD 3-1 Reading	ACRS
OGC	GGrant, RIII	GHill (2)	

+See previous concurrence
*See memo F. Reinhart to L. Raghavan, dated 12/04/02

ADAMS Accession Number: ML030210062

OFFICE	PM:PD3-1	LA:PD3-1	SC:SPSB	SC:SPLB	OGC	SC:PD3-1
NAME	JLamb	THarris	FMReinhart*	EWeiss	SUttal+	LRaghavan
DATE	03/11/03	03/13/03	12/04/02	03/12/03	3/10/03	03/12/03

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 166
License No. DPR-43

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

   A. The application for amendment by Nuclear Management Company, LLC (NMC or the licensee), dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002, 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, by Amendment No. 166 Facility Operating License No. DPR-43 is hereby amended to revise the current radiological consequence analyses for the KNPP design-basis accidents to implement the alternate source term (AST) as described in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors" and pursuant to 10 CFR 50.67, "Accident Source Term" as set forth in the license amendment application dated March 19, 2002, supplemented by letters dated September 13 and October 21, 2002, and evaluated in the associated safety evaluation by the Commission's Office of Nuclear Reactor Regulation.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Date of Issuance: March 17, 2003

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 166 TO FACILITY OPERATING LICENSE NO. DPR-43

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

By letter dated March 19, 2002, as supplemented by letters dated September 13 and October 21, 2002, Nuclear Management Company, LLC (NMC or the licensee), requested revisions to the current radiological consequence analyses for the Kewaunee Nuclear Power Plant (KNPP) design-basis accidents (DBAs). The licensee proposed to implement the alternative source term (AST) in this revision as described in Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors” and pursuant to 10 CFR 50.67, “Accident Source Term.” The current Kewaunee AST was developed using Technical Information Document (TID)-14844, “Calculation of Distance Factors for Power and Test Reactor sites.” The licensee has not requested any changes to the current KNPP Technical Specifications or any modifications to the plant design at this time.

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

The proposed revisions would revise the current KNPP radiological consequence analyses for the following eight (8) DBAs described in Section 14 of the KNPP Updated Safety Analysis Report (USAR):

1. Loss-of-Coolant Accident (LOCA)
2. Steam Generator Tube Rupture (SGTR) Accident
3. Main Steamline Break (MSLB) Accident
4. Fuel Handling Accident (FHA)
5. Locked Rotor Accident (LRA)
6. Rod Ejection Accident (REA)
7. Gas Decay Tank Rupture (GDTR)
8. Volume Control Tank Rupture (VCTR)

ENCLOSURE

2.0 REGULATORY EVALUATION

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. The current radiological consequence analyses for the DBAs for KNPP are based upon the TID-14844 Accident Source Term. In 1995, the Nuclear Regulatory Commission (NRC) staff published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." NUREG-1465 provides estimates of the AST that were more physically based and that could be applied to the design of future light-water power reactors. NUREG-1465 presents a representative AST for a boiling-water reactor and for a pressurized-water reactor. These source terms are characterized by the composition and magnitude of the radioactive material, the chemical and physical properties of the material, and the timing of the release to the containment.

The NRC staff considered the applicability of the revised source terms in NUREG-1465 to operating reactors and determined that the current analytical approach based on the TID-14844 source term would continue to be adequate to protect public health and safety. Operating reactors licensed under that approach would not be required to re-analyze accidents using the revised source terms. The NRC staff also determined that some licensees might wish to use an AST in their analyses to support cost-beneficial licensing actions. The NRC staff, therefore, initiated several actions to provide a regulatory basis for operating reactors to use an AST in design-basis radiological consequence analyses. These initiatives resulted in the development and issuance of 10 CFR 50.67 and RG 1.183. As part of the implementation of the AST, the total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67 replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 as follows:

	10 CFR 50.67 GDC 19	10 CFR 100.11 GDC 19
Exclusion Area Boundary and Low Population Zone	25 rem TEDE	300 rem thyroid and 5 rem whole body
Control Room	5 rem TEDE	5 rem whole body, or its equivalent to any part of the body

A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67 to voluntarily revise its current AST used in design-basis radiological consequence analyses for a license amendment under 10 CFR 50.90. In this license amendment, the licensee requested a full-scope implementation of the AST, as described in RG 1.183 pursuant to 10 CFR 50.67 for changing the KNPP design-basis radiological consequence analyses for the DBAs. In general, information provided by RG 1.183 is reflected in Chapter 15.0.1 of the Standard Review Plan (SRP), “Radiological Consequence Analyses Using Alternative Source Term.”

Other relevant regulatory requirements applicable to this license amendment are (1) GDC 19, “Control Room” of Appendix A to 10 CFR Part 50, and NUREG-0737 III.D.3.4 as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation, and toxic gases.

3.0 TECHNICAL EVALUATION

The licensee re-analyzed and submitted the radiological consequence analyses for 8 affected DBAs. The NRC staff requested in its Request for Additional Information (RAI), dated July 3, 2002, that the licensee provide the detailed radiological dose calculations complete with copies of the inputs prepared and outputs obtained from the computer code used in the dose calculations. In a response to the request, the licensee stated that the licensee’s contractor (Westinghouse) does not provide copies of proprietary dose calculation notes for NRC review, and instead, proposed to bring the dose calculation notes to the Westinghouse office in Rockville, MD for NRC review. The NRC staff accepted the licensee’s proposal and reviewed the following Westinghouse proprietary dose calculation notes provided at the Westinghouse office in Rockville, MD. The NRC staff confirmed that the methods used in these notes for dose calculations are consistent with the guidelines provided in Regulatory Guide 1.183 except as noted in this safety evaluation. These documents were prepared by Westinghouse in support of the licensee’s proposed revision to the KNPP design-basis radiological consequence analyses:

1. CN-CRA-99-01, Revision 1, “Kewaunee Steam Releases for Radiological Dose Calculations.”
2. CN-CRA-99-036, Revision 2, “Kewaunee SGTR Offsite Radiation Dose Analysis for Replacement Steam Generator Program.”
3. CN-CRA-00-69, Revision 0, “Kewaunee Main Steamline Break Doses using AST Methodology.”
4. CN-CRA-99-40, Revision 0, “Kewaunee Determination of Iodine Spray Removal Coefficients and DF Limits.”
5. CN-CRA-00-56, Revision 0, “Kewaunee AST FHA Doses.”
6. CN-CRA-99-29, Revision 1, “Kewaunee SGTR T&H Analysis for Replacement Steam Generator Program.”
7. CN-CRA-99-28, Revision 0, “Kewaunee Definition of Iodine Spike Rate and Duration.”
8. CN-CRA-99-46, Revision 2, “Kewaunee GDTR and VCTR Radiation Dose Analyses for Replacement Steam Generator Program.”
9. CN-CRA-00-2, Revision 1, “Kewaunee AST LOCA Doses.”
10. CN-CRA-00-68, Revision 0, “Kewaunee AST RCA Doses
11. CN-CRA-00-70, Revision 2, “Kewaunee AST REA Doses.”

The NRC staff considers the implementation of the AST to be a significant change to the KNPP design-basis. In order to accept the use of an AST and the TEDE dose criteria, the NRC staff must make a current finding of compliance with regulations applicable to this request. However, the NRC staff accepted prior KNPP design-basis assumptions or parameters that are unrelated to the use of the AST, or are unaffected by the AST and they were allowed to be continued as the licensee’s design-basis. The radiological consequence analyses are based on a reactor core power of 1650 MWt increased to 1683 MWt to cover uncertainty. The licensee conservatively increased the reactor core fission product inventory and the reactor coolant fission product concentrations by an additional 10 percent to allow future KNPP power uprating.

3.1 Loss-of-Coolant Accident

The current radiological consequence analysis for the postulated LOCA is based on the TID-14844 source term and it is provided in the KNPP USAR Section 14.3. To demonstrate that the engineered safety features (ESFs) systems designed to mitigate the radiological consequences following the postulated LOCA at Kewaunee will remain adequate after implementation of the AST, the licensee re-analyzed the offsite and control room radiological consequences of the postulated LOCA.

The licensee provided the results of its offsite and control room dose calculations and the major assumptions and parameters used in its dose calculations. As documented in its submittals, the licensee has determined that after implementation of the AST, the existing ESF systems at KNPP will provide assurance that the total radiological consequences of the postulated LOCA at the exclusion area boundary (EAB), in the low population zone (LPZ), and in the control room will meet the acceptable radiation dose criteria specified in 10 CFR 50.67 (b)(2). As part of the implementation of the AST, the TEDE acceptance criterion of 10 CFR 50.67 (b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 and whole body dose guideline of GDC 19.

The NRC staff has reviewed the licensee's analyses and has performed an independent confirmatory radiological consequence dose calculation for the following three potential fission product release pathways after the postulated LOCA:

(1) containment leakage pathway,
(2) leakage pathway from ESF systems outside containment, and
(3) emergency core cooling system (ECCS) Recirculation back-leakage pathway to the Refueling Water Storage Tank (RWST).

3.1.1 Containment Leakage Pathway

The current KNPP containment design-basis leak rate is 0.5 weight percent per day. For the radiological consequence analysis, this rate is followed by 0.25 percent per day after 24 hours following a LOCA for the entire duration of the accident (30 days) consistent with the guidance provided in RG 1.183. Immediately following the accident, the shield building pressure increases due to heat transferred from the containment shell. Operation of one of two shield building ventilation fans will establish a negative pressure within the shield building. For the first 10 minutes into the postulated LOCA, it is assumed that 90 percent of the containment leakage is released directly to the environment without holdup or filtration. The licensee used this 10 minutes in its dose calculation instead of 6 minutes as described in the current KNPP USAR. The NRC staff finds this change to be acceptable since the 10 minute is more conservative. The remaining 10 percent of the containment leakage is assumed to enter the auxiliary building where it is filtered by the auxiliary building special ventilation system prior to release to the environment. These leakage assumptions are specified in the KNPP Technical Specification (TS) Section 6.20, "Containment Leakage Rate Testing Program," and described in more detail in Section 14.3.5 of the KNPP USAR.

Following achievement of a vacuum in the shield building at 10 minutes into the accident, only 1 percent of the containment leakage is assumed to be released directly to the environment without holdup or filtration, 10 percent continues to enter the auxiliary building where it is still filtered by the auxiliary building special ventilation system (ABSVS) prior to release to the environment, and the remaining 89 percent is assumed to enter the shield building where it is filtered by the shield building ventilation system (SBVS) prior to release to the environment. These leakage assumptions are also specified in the KNPP TS Section 6.20, “Containment Leakage Rate Testing Program,” and described in more detail in Section 14.3.5 of the Kewaunee USAR.

Three time periods are associated with the shield building ventilation operation. During the first period (0 to 10 minutes), the SBVS starts and draws a vacuum in the shield building. During this period, the licensee assumed no credit for the shield building. The second period (10 to 30 minutes), the SBVS dampers will modulate to maintain a vacuum. During this period, the licensee assumed that the containment leakage (89 percent) will be processed by the SBVS filters prior to release to the environment. No credit is taken for recirculation. The final period (greater than 30 minutes) consists of stable system operation with a combination of recirculation and discharge to the environment to maintain the vacuum in the shield building. During this final period, the licensee has taken fission product removal credit by the SBVS filters for recirculation as well as release to the environment.

The fission products in the containment atmosphere following the postulated LOCA is mitigated by natural deposition processes and by the containment spray system (CSS). The licensee assumed a radioactive aerosol removal rate of 0.1 per hour in the containment atmosphere. This removal credit is taken after the CSS operation is terminated. The NRC staff finds 0.1 per hour aerosol removal rate to be reasonable (within the 85 percent of the uncertainty distribution) based on theits study published in NUREG/CR-6189, “A Simplified Model of Aerosol Removal by Natural Processes in Reactor containment,” and is therefore, acceptable.

The KNPP CSS is an ESF system and is designed to provide reactor building cooling and fission product removal in the containment following the postulated LOCA. The CSS consists of two spray pumps and they are automatically started by the coincidence of three sets of one out of two high-high containment pressure signals. The licensee assumed that one train of the CSS will operate following the postulated LOCA taking suction initially from the RWST with no startup delay until the water in the RWST reaches a pre-set low level in 0.91 hours after the accident. The spray pump suction is then transferred manually to the containment sump and the spray water is re-circulated. The licensee assumed fission product removal by the CSS during only initial spray operation and conservatively assumed no fission product removal during recirculation phase. The CSS is assumed to terminate at 0.91 hours.

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 by the CSS for iodine in elemental form and fission products in particulate form. The NRC staff finds that the removal rates calculated by the licensee are acceptable. The major parameters and assumptions used by the licensee including the spray removal rates are listed in Table 2. The containment leakage pathway contributed the most radiological consequence dose (greater than 97 percent) at the EAB, in the LPZ, and control room (see Table 1).

3.1.2 Post-LOCA Leakage Pathway From Engineered Safety Features Outside Containment

Any leakage water from ESF components located outside the containment releases fission products during the recirculating phase of long-term core cooling following a postulated LOCA. The leakage from ESF components occurs in the auxiliary building. The licensee conservatively assumed the leakage into the auxiliary building to start from the start of the accident. The leakage rate is assumed to be 6 gallons per hour. The licensee assumed that 1 percent of the total iodine activity in the leaked fluid becomes airborne and is released through the ABSVS to the environment during the entire period of the accident (30 days). The licensee further assumed that half of the iodine activity that becomes airborne from leak sources in the auxiliary building.

The iodine partition factor of 1 percent is a departure from the guidance provided in RG 1.183 which states that an iodine partition factor of 10 percent should be assumed unless a smaller amount can be justified based on the actual sump water pH history and area ventilation rate. The KNPP USAR Section 6.2.5, “Effects of Leakage from Residual Heat Removal System,” identifies that the temperature of the containment recirculation water is below 212 °F when ECCS recirculation begins and that any leakage liquid is therefore assumed to be sub-cooled and to remain liquid retaining iodine contained in any leakage water in the liquid.

The licensee stated that 1 percent iodine partition factor and 50 percent iodine plateout assumptions were contained in the original KNPP USAR Section 14.3.7 (original page 14.3-68) which were approved by the Atomic Energy Commission. Therefore, the licensee considers that these assumptions are parts of Kewaunee original license and that they are still current licensing bases. Furthermore, the NRC staff believes that this assumption is unrelated to the use of the AST. Therefore, the NRC staff finds that 1 percent iodine partition factor and 50 percent iodine plateout assumptions are acceptable. The radiological consequence contribution from this pathway is insignificant (less than 1 percent) at the EAB and to the control room operator (see Table 1).

3.1.3 Emergency Core Cooling System Back-Leakage Pathway to Refueling Water Storage Tank

Following a postulated LOCA, the suction water source for the ECCS is switched from the RWST to the containment sump. The leakage path back to the RWST is through the suction lines of the safety injection, internal containment spray, and residual heat removal pumps. The licensee conservatively assumed the leakage into the RWST to start from the start of the accident and a back-flow leakage of 3 gallons per minute (gpm). The licensee stated in their response dated August 23, 2002, to the NRC staff’s RAI that the back-leakage to the RWST is ensured not to exceed 3 gpm through a combination of visual inspections and hydraulic tests in accordance with KNPP Surveillance Procedure.

The licensee also assumed 1 percent iodine partition in the RWST leakage water as it did for the ECCS leakage to the auxiliary building. The licensee stated that in addition to this leakage water being below 212 °F, it is leaked into RWST water that will be significantly below 212 °F. The leakage water enters the RWST at the bottom of the tank. The NRC staff finds that the leakage rate and iodine partition factor assumed by the licensee are acceptable. The radiological consequence contribution from this pathway is less significant (less than 2 percent) at the EAB for the postulated LOCA (see Table 1).

3.1.4 Radiological Consequence of Loss-of-Coolant Accident

The licensee reevaluated the radiological consequence resulting from the postulated LOCA using the AST and concluded that the radiological consequences at the EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff has reviewed the AST implementation proposed by the licensee. In performing this review, the NRC staff relied upon information provided by the licensee, NRC staff experience in performing similar reviews, and where deemed necessary, on NRC staff's confirmatory calculations.

To verify the licensee's radiological consequence analyses, the NRC staff performed a confirmatory radiological consequence dose calculation. The results were also within the dose criteria specified in 10 CFR 50.67. Although, the NRC staff performed its independent radiological consequence dose calculations, as a means of confirming the licensee's results, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence dose calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and the NRC staff are listed in Tables 2 through 9. The radiological consequences at the EAB, and the LPZ, and in the control room, as calculated by the licensee and by the NRC staff, are all within the dose criteria specified in 10 CFR 50.67. Therefore, the NRC staff concludes that the proposed AST implementation revising the current design-basis radiological consequence analysis for the postulated LOCA is acceptable.

The bases for the NRC staff's acceptance are that the radiological consequences analyzed by the licensee at EAB, LPZ, and in the control room are within the dose criteria specified in 10 CFR 50.67.

3.2 Fuel Handling Accident

The current radiological consequence analysis for the postulated design-basis FHA is based on the accident source term described in TID-14844 and it is provided in KNPP USAR Section 14.2.1. The licensee re-evaluated the radiological consequences of a postulated FHA in the containment with no credit taken for containment isolation implementing the AST. Since the assumptions and parameters used for a FHA inside containment are identical to those for a FHA in the auxiliary building, the resulting radiological consequences are the same regardless of the location of the accident.

The licensee concluded in the submittals that the radiological consequences resulting from the postulated FHA in the containment with no credit taken for containment isolation are within the dose acceptance criteria specified in SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and GDC 19.

The licensee reached this conclusion as a result of:

(1) using the guidance provided in Appendix B to RG 1.183, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident,"

(2) taking no credit for containment isolation,

(3) taking no credit for removal of fission products by the spent fuel pool ventilation system in the auxiliary building,

(4) using an overall decontamination factor of 200 for iodine in elemental and particulate forms in the spent fuel pool water with minimum water depth of 23 feet consistent with the guidelines provided in RG 1.183,

(5) releasing all fission products within 2 hours using an exponential release model with higher release in the initial period,

(6) assuming all fuel rods in one fuel assembly with an axial power peaking factor of 1.7 are damaged to the extent that the entire gap activity inventory of the damaged fuel rods is released to the surrounding water,

(7) using a fission product decay period of 100 hours (time period from the reactor shutdown to the first fuel movement).

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated FHA. The radiological consequences calculated by the NRC staff are well within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1 (6.3 rem TEDE at the EAB).

Even though the NRC staff performed its confirmatory dose calculations, the NRC staff's acceptance is based on the licensee's analyses. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 3. The radiological consequences at the EAB, and the LPZ, and in the control room as calculated by the licensee are all well within the dose criterion specified in GDC 19 and meet the dose acceptance criterion specified in the SRP 15.0.1. Therefore, the NRC staff concludes that the proposed AST implementation revising the current design-basis radiological consequence analysis for the postulated FHA is acceptable.

3.3 Steam Generator Tube Rupture Accident

This DBA postulates a rupture in a tube in one of the two steam generators resulting in the transfer of reactor coolant water to the ruptured steam generator. The primary-to-secondary flow through the ruptured tube (break flow) following a SGTR results in a depressurization of the reactor coolant system (RCS), a reactor trip, and actuation of safety injection. After safety injection actuates, it is assumed that the RCS pressure will stabilize at a value at which the safety injection and break flows are equal. The break flow is assumed to continue until plant operators have taken action to reduce RCS pressure. When RCS pressure is less than the steam generator (SG) pressure, the pressure differential and the flow direction reverses, terminating the break flow. The licensee assumed this occurs within 30 minutes from safety injection actuation.

At 8 hours after the accident, the licensee assumed the residual heat removal system (RHR) will be placed into service for heat removal and there will be no further steam release to the environment from the secondary system. Break flows, steam releases, and feedwater flows were determined using thermal-hydraulic analyses to bound the operating conditions. The analysis assumed that the noble gases entrained in the break flow are released to the environment without holdup or decontamination in the SGs. For the ruptured SG, the analysis assumed that part of the break flow will immediately flash to steam and the entrained gases be released to the environment with no holdup in the SGs. The portion of the break flow that does not flash is assumed to mix with the bulk water in the SGs and be released at the steaming rate of the SGs. The iodine release rates are reduced to account for partitioning between the liquid and vapor phases. The licensee provided two partitioning factors at pre-reactor trip and post-reactor trip. While the flash fraction would be greater before the trip, the associated releases would be via the main condenser which would afford iodine mitigation.

The licensee assumed the initial iodine inventory in the RCS and SG to be at the maximum concentrations permitted by TSs. The initial noble gas inventory in the RCS is based on fuel damage equivalent to 1.0 percent failed fuel. Two iodine spiking cases are considered. The first assumes that an iodine spike occurred just before the SGTR and that the RCS iodine inventory is at 60 μCi/gm dose equivalent I-131. The second case assumes the event initiates an iodine spike. In this case, iodine is released from the fuel to the RCS at a rate of 500 times the normal iodine appearance rate. This multiplier is more conservative than the 335 times multiplier specified by RG 1.183, and therefore, it is acceptable.

The licensee determined the iodine appearance rates assuming a letdown system flow rate of 88 gpm, radioactive decay, a primary system leakage of 12 gpm, and complete removal of iodine by the letdown system demineralizer (most conservative for the purpose of determining the iodine appearance rates). The NRC staff has verified the iodine appearance rates provided by its own calculations and finds that they are within the guidelines provided within Regulatory Guide 1.183. The licensee stated that the iodine release rate is such that the iodine inventory of the fuel rod gap will be depleted by four hours. In response to NRC staff questions, the licensee provided additional information on this assumption in their August 23, 2002, letter. The licensee assumed that the spike duration is a function of the accident-induced iodine appearance rate (148.6 Ci/min for iodine-131) and the iodine available for release from the fuel gap of fuel rods with defects. This evaluation yields a release duration of 3.4 hours. The NRC staff finds the licensee's justification of the 4-hour spike duration acceptable.

The licensee re-evaluated the radiological consequence resulting from the postulated SGTR accident and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose acceptance criteria specified in SRP 15.0.1. The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated SGTR accident. The radiological consequences calculated by the NRC staff are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1. The radiological consequences calculated by the NRC staff are within 5 percent of those calculated by the licensee. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 4.

3.4 Main Steamline Break Accident (MSLB)

This DBA postulates an unisolable failure in one of the two main steam lines at a location outside of containment, resulting in the release of steam from the affected steam line. The faulted SG will rapidly depressurize and release its entire liquid inventory and dissolved iodine through the faulted steam line to the environment. The rapid secondary depressurization causes a reactor power transient, resulting in a reactor trip. The released steam may be contaminated due to leakage of reactor coolant into the SGs via small tube leaks (i.e., primary-to-secondary leakage). The radiological consequences of a break outside containment will bound those results from a break inside containment. Thus, only the break outside containment is analyzed.

The licensee assumed that the faulted SG boils dry in two minutes and that primary-to-secondary leakage transfers 500 gallons per day (gpd), bounding the KNPP TS limit of 150 gpd per SG. This leakage mixes with the bulk SG water. Transferred noble gases are released without a holdup. Iodine is released to the environment at the steaming rate of the SGs with credit for iodine partitioning. The licensee assumed that the RHR system will be available for heat removal at 8 hours after the accident and that after 8 hours, there will be no further steam releases to the environment from the intact SG. The licensee further assumed that within 72 hours after the accident, the RCS has been cooled to below 212 °F, and there will be no further steam release from the faulted SG. The licensee's assumptions regarding the RCS inventory and iodine spiking are the same as those discussed above for the SGTR. No fuel damage is projected for the MSLB.

The licensee re-evaluated the radiological consequence resulting from the postulated MSLB accident and concluded that the radiological consequences at the EAB, LPZ and in the control room are within the dose criteria specified in 10 CFR 50.67. The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological dose consequence analyses and finds that they are consistent with the conservative guidance provided in RG 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated MSLB accident. The radiological consequences calculated by the NRC staff are well within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1. The radiological consequences calculated by the NRC staff are within 10 percent of those calculated by the licensee. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 5.

3.5 Locked Rotor Accident

For this DBA, a reactor coolant pump rotor is assumed to seize instantaneously causing a rapid reduction in the flow through the affected RCS loop. A reactor trip will occur, shutting down the reactor. The flow imbalance creates localized temperature and pressure changes in the core. If severe enough, these differences may lead to localized boiling and fuel damage. The radiological consequences are due to leakage of the contaminated reactor coolant to the SGs and then releases from the SGs to the environment.

The licensee conservatively assumed that all of the fuel rods in the core are damaged and that all of its fuel gap activity is released to the reactor coolant system as a result of the primary coolant pump locked rotor accident. The NRC staff finds this assumption to be conservative based on the NRC staff's experience in performing similar reviews for other reactor plants. At 8 hours after the accident, the licensee assumed the RHR system will be placed into service for heat removal and there will be no further steam release to the environment from the secondary system. All other parameters and assumptions for fission product transport and release mechanisms are same as those discussed in the above Section 3.3, "SGTR Accident."

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological consequence analyses and finds that they are consistent with the guidance provided in Regulatory Guide 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated locked rotor accident.

The radiological consequences calculated by the licensee and by the NRC staff are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criterion specified in the SRP 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 6.

3.6 Rod Ejection Accident

This DBA postulates the mechanical failure of a control rod drive mechanism pressure housing that results in the ejection of a rod cluster control assembly and drive shaft. Localized damage to fuel cladding and a limited amount of fuel melt are projected.

The release to the environment is assumed to occur through two pathways:

- Release of containment atmosphere (i.e., design leakage)
- Release of RCS inventory via primary-to-secondary leakage through SGs

The licensee assumed that 15 percent of the fuel rods suffer sufficient damage to result in the release of all of their gap inventory to the RCS or to the containment. The licensee further assumed that 10 percent of the core inventory of noble gases and iodine are in the fuel rod gap, consistent with the guideline provided in RG 1.183. A small fraction of the fuel in the failed rods is assumed to melt because of the accident. The licensee estimated this damage to be limited to 0.375 percent of the core. This is based on the assumption that 50 percent of the rods in departure from nuclear boiling undergo centerline melting and the melting is limited to the inner 10 percent occurring over 50 percent of the axial length of the affected rods. The NRC staff finds these assumptions to be conservative and therefore, acceptable.

The licensee assumed that 100 percent release of noble gases and 50 percent of iodine to the RCS, and 100 percent release of noble gases and 25 percent to the containment which are consistent with the guidelines provided in RG 1.183. The licensee conservatively assumed that a pre-accident iodine spike occurred just before the event such that the RCS specific activity has a concentration of 60 µCi/gm dose equivalent I-131.

For the containment leakage case, the iodine released is 95 percent cesium iodide, 4.85 percent elemental, and 0.15 percent organic. The licensee assumed the iodine species in the secondary system to be 97 percent elemental and 3 percent organic. The containment is projected to leak at its design leakage of 0.5 percent of its contents by weight per day for the first 24 hours and then at 0.25 percent for the remainder of the 30-day accident duration. These assumptions are also consistent with the guidelines provided in RG 1.183. For the containment leakage pathway, the licensee did not take credit for fission product removal by spray or aerosol deposition.

The NRC staff reviewed the licensee's methods, parameters, and assumptions used in its radiological consequence analyses and finds that they are consistent with the guidance provided in Regulatory Guide 1.183. To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological consequence dose calculations for the postulated rod ejection accident.

The radiological consequences calculated by the licensee and by the NRC staff for the rod ejection accident are within the dose criterion specified in GDC 19 (5 rem TEDE in the control room), and meet the dose acceptance criteria specified in the SRP 15.0.1. The results of the licensee's radiological consequence calculations are provided in Table 1 and the major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 7.

3.7 Gas Decay Tank and Volume Control Tank Rupture Accidents

The KNPP licensing basis includes analyses of the radiological consequences of a rupture of a gas decay tank (GDT) and a rupture of the volume control tank (VCT). The current SRP does not classify these accidents as a DBA. The GDTs are used to store processed radioactive gases removed from the reactor coolant to allow for radioactive decay before the controlled release to the environment. The VCT is a component in the plants' chemical and volume control systems that serves as a surge volume to balance differences in letdown and makeup flow rates while maintaining reactor coolant inventory. Part of the reactor coolant (known as letdown) is removed from the RCS, cooled, filtered, demineralized, and degassed.

The GDT rupture case assumes that the entire inventory of gases in the RCS based on continuous operation with 1.0 percent failed fuel is in the GDT and is released over a period of five minutes. The initial inventory of noble gases in the VCT is based on continuous operation with 1.0 percent failed fuel, without purging of the VCT. The initial inventory of iodine is based on continuous operation with an RCS specific inventory of 60 µCi/gm dose equivalent I-131, with a 90 percent removal by the letdown demineralizer. The initial inventory is assumed to be released over a period of five minutes. After the event starts, letdown flow to the VCT is assumed to continue at the maximum flow rate of 88 gpm for 5 minutes, after which letdown is isolated. Iodine in the letdown flow are reduced by 90 percent by the letdown demineralizer.

To verify the licensee's radiological consequence assessments, the NRC staff performed confirmatory radiological dose calculations for the postulated GDT and VCT rupture accidents.

The radiological consequences calculated by the licensee and by the NRC staff for the GDT rupture and VCT rupture are small fractions (less than 1 percent) of the dose criteria specified in 10 CFR 50.67 and therefore, the NRC staff finds them acceptable. The major parameters and assumptions used by the licensee and acceptable to the NRC staff are listed in Table 8.

3.8 Control Room Habitability

The KNPP control room habitability systems include radiation shielding, two control room air conditioning units, redundant control room post-accident recirculation filter/fan units, control room isolation, radiation monitoring, and redundant outside air intakes. During normal operation, one air conditioning unit recirculates control room air at 13,450 cubic feet per minute (cfm). This air is mixed with 2,500 cfm of unfiltered outside makeup air.

Upon a safety injection signal and/or high radiation detection, the control room is automatically isolated, 100 percent of control room air is recirculated, and 2,500 cfm of the recirculation flow is processed through one of two control room post-accident recirculation filter/fan units.

Each control room post-accident recirculation filter/fan unit consists of, among other things, a pre-filter, high efficiency particulate air filter, and 2-inch deep charcoal adsorber.

In 1989, the licensee submitted an updated control room habitability evaluation report in response to the NRC concerns expressed in NUREG/CR-4960, "Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations (1988)." The NUREG/CR-4960 presented the results of a survey of control room habitability systems at 12 nuclear power plants. In its updated control room habitability evaluation report, the licensee documented actions taken to address the NRC concerns including unfiltered air inleakage into the control room.

The licensee stated in the report that they performed air flow measurements in the system using an electronic micro-manometer to determine the unfiltered air inleakage into the control room envelope and the system performance. The licensee did not performed an integrated control room unfiltered air inleakage test. Instead, the licensee estimated the unfiltered air inleakage to be 200 cfm based on the air flow measurements. The NRC staff accepted the licensee's report and 200 cfm unfiltered air inleakage rate assumption in both KNPP License Amendment No. 88 issued in 1990 and in KNPP License Amendment No. 132 issued in 1997. The licensee considers the 200 cfm as the current KNPP licensing and design bases.

In its response dated September 13, 2002, to the NRC staff's RAI, the licensee stated that the unfiltered inleakage was determined by using the measured leakage through the closed dampers (48 cfm), allowance for leakage through building elements (80 cfm), and air exchange based on door opening and closing (10 cfm). The leakage was adjusted based on the worse case leakage resulting from one of the redundant dampers failing to close. The licensee stated that this resulted in a total leakage of 200 cfm.

The NRC staff is currently working toward a resolution of generic issues related to control room habitability, with particular focus on the validity of the control room unfiltered air inleakage rates that are commonly assumed in licensee's analyses of control room habitability. The NRC staff's acceptance of the 200 cfm unfiltered air inleakage assumption in this license amendment does not preclude any future generic regulatory actions that may become applicable to KNPP.

This assessment may be used in subsequent amendments; however, any use of this assessment that involves a relaxation in requirements will require verification (in accordance with the aforementioned resolution of the generic issues related to control room habitability) that the unfiltered inleakage rate is within limits assumed in the AST assessment.

The NRC staff has determined that there is reasonable assurance that the KNPP control will be habitable with up to 200 cfm unfiltered air inleakage during the postulated DBAs and this amendment may be approved before the resolution of control room generic issue. The NRC staff bases this determination on (1) the maximum allowable unfiltered air inleakage rate of up to 200 cfm meeting the relevant dose acceptance criteria specified in 10 CFR 50.67 and GDC 19, (2) conservative assumptions and parameters used in the radiological consequence analyses, and (3) the low probability of the postulated accidents, occurring during this interim period until the NRC staff resolves this generic issue, that could result in radioactivity releases sufficient to challenge the ability of control room operators to protect the health and safety of the public.

3.9 Technical Conclusion

The NRC staff has reviewed the licensee's analyses and performed confirmatory assessments of the radiological consequence of the postulated DBAs. The doses calculated by the licensee are listed in Table 1. The doses calculated by the licensee and by the NRC staff are all within relevant dose criteria specified in 10 CFR 50.67 and SRP 15.0.1. Therefore, the NRC staff concludes that the radiological consequences analyzed and submitted by the licensee are acceptable. The NRC staff further concludes that the implementation of the AST replacing the current KNPP TID-14844 source term for the KNPP DBAs identified in the licensee's submittal is 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 (67 FR 18646). 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 NRC 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.

Principal Contributor: J. Lee

Date: March 17, 2003

Table 1
Radiological Consequences Expressed as TEDE
(rem)

Design-basis Accidents	EAB[1]	LPZ[2]	Control Room
LOCA			
Containment leakage pathway	1.26	0.221	4.35
ECCS leakage pathway	1.44E-3	1.16E-3	2.79E-2
RWST leakage pathway	4.33E-2	1.23E-2	5.66E-1
Total (rounded up)	1.31	0.24	4.95
Dose acceptance criteria[3]	25	25	5.0
Fuel handing accident	0.6	0.11	1.0
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]
Steam generator tube rupture			
Pre-accident iodine spike	1.3	0.3	3.0
Dose acceptance criteria[3]	25	25	5.0
Accident-initiated iodine spike	0.8	0.2	1.0
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Main steamline break			
Pre-accident iodine spike	0.05	0.02	0.7
Dose acceptance criteria[3]	25	25	5.0
Accident-initiated iodine spike	0.2	0.05	2.3
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Locked rotor accident	1.7	0.3	4.3
Dose acceptance criteria	2.5[4]	2.5[4]	5.0[3]
Control rod ejection accident	0.52	0.11	1.9
Dose acceptance criteria	6.3[4]	6.3[4]	5.0[3]

(1) Exclusion area boundary
(2) Low population zone
(3) 10 CFR 50.67
(4) SRP 15.0.1

Table 2
Parameters and Assumptions Used in
Radiological Consequence Calculations
Loss-of-Coolant Accident

Parameter	Value
Reactor power	1851.3 MWt
Containment volume	1.32E+6 ft^3
Containment leak rates	
0 to 1 hour	0.5 percent per day
1 to 720 hours	0.25 percent per day
Aerosol removal rate (after 0.91 hours)	0.1 per hour
Iodine removal rates by spray (per hour)	
0 to 0.91 hours	
Elemental	20
Particulate	5
0.91 to 720 hours	
Elemental	0
Particulate	0
Containment sump volume	4.21E+4 ft^3
ECCS leak rate	
0 to 720 hours	6 gph
Iodine partition factor	1 percent
ECCS leak rate to RWST	
0 to 24 hours	3.0 gpm
24 to 720 hours	1.5 gpm
Iodine partition factor	1 percent
Control room	
Volume	1.276E+5 ft^3
Filtered makeup air flow	0
Filtered Recirculation air flow	2500 cfm
Unfiltered air inleakage rate	200 cfm
Filter efficiencies	
Aerosol	99 percent
Elemental iodine	90 percent
Organic iodine	90 percent

Table 3
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Fuel Handling Accident

Parameter	Value
Reactor power	1851.3 MWt
Radial peaking factor	1.7
Fission product decay period	100 hours
Number of fuel assembly damaged	1
Fuel pool water depth	23 ft
Fuel gap fission product inventory	
Noble gases excluding Kr-85	5 percent
Kr-85	10 percent
I-131	8 percent
Alkali metals	12 percent
Fuel pool decontamination factors	
Iodine	200
Noble gases	1
Duration of accident	2 hours

Table 4
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Steam Generator Tube Rupture Accident

Reactor coolant activity	
Accident-initiated iodine spike case	1.0 µCi/gm dose equivalent I-131
Pre-accident iodine spike case	60.0 µCi/gm dose equivalent .I-131
Primary-to-secondary break flow	
Pre-trip	1.68E+4 lbm
Post-trip	1.38E+5 lbm
Break flow flash fractions	
Pre-trip	19.71
Post-trip	14.76
Break flow duration	30 minutes
Steam generator mass	2.685E+5 lbm
Steam release from ruptured steam generator	8.5E+4 lbm (post-trip)
Steam release from intact steam generator	
0-2 hours	2.438E+5 lbm
2-8 hours	5.066E+5 lbm
Intact steam generator iodine activity	0.1 µCi/gm dose equivalent .I-131
Steam iodine partition coefficient	0.01
Intact steam generator tube leak rate	500 gallons per day
Iodine spiking factor	500
Loss of offsite power	At reactor trip
Reactor trip time	173.7 seconds

Table 5
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Main Steam Line Break Accident

Reactor coolant activity	
Accident-initiated iodine spike case	1.0 µCi/gm dose equivalent I-131
Pre-accident iodine spike case	60.0 µCi/gm dose equivalent .I-131
Steam generator mass	2.685E+5 lbm
Steam release from faulted steam generator	1.5E+5 lbm
Time to release initial mass in faulted steam generator	2 minutes
Steam release from intact steam generator	
0-2 hours	2.2E+5 lbm (+10 percent)
2-8 hours	3.93E+5 lbm (+10 percent)
Intact steam generator activity	0.1 µCi/gm dose equivalent .I-131
Steam partition coefficient	
Faulted steam generator	1
Intact steam generator	0.01
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine spiking factor	500
Duration of iodine spike	4 hours

Table 6
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Locked Rotor Accident

Reactor coolant activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Secondary coolant activity prior to accident	0.1µCi/gm dose equivalent .I-131
Fraction of fuel rods failed	100 percent
Gap fractions	Per Regulatory Guide 1.183
Steam release from steam generator	
0-2 hours	2.13E+5 lbm (+10 percent)
2-8 hours	4.18E+5 lbm (+10 percent)
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine partition factor in steam generator	0.01

Table 7
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Rod Ejection Accident

Reactor coolant activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Secondary coolant activity prior to accident	0.1µCi/gm dose equivalent .I-131
Fraction of fuel rods failed	15 percent
Gap fractions(percent)	
Iodine	10
Noble gas	10
Alkali metals	12
Fraction of fuel melted (percent)	0.375
Fraction activity released from melted fuel (percent)	
For containment leakage pathway	
Iodine	25
Noble gas	100
Alkali metals	100
For primary to secondary leakage pathway	
Iodine	50
Noble gas	100
Alkali metals	100
Steaming rates (lbm per second)	
0-200 seconds	800
222 -1800 seconds	100
After 1800 seconds	0
Steam generator tube leak rate	500 gallons per day per steam generator
Iodine partition factor in steam generator	0.01
Containment leak rates	
0 to 1 hour	0.5 percent per day
1 to 720 hours	0.25 percent per day
Aerosol removal rate in containment	0

Table 8
Parameters and Assumptions
Used in
Radiological Consequence Calculations
Gas Decay Tank and Volume Control Tank Ruptures

Reactor coolant iodine activity prior to accident	60.0 µCi/gm dose equivalent .I-131
Reactor coolant noble gas activity prior to accident	1 percent fuel defect
Release duration (minutes)	
Gas decay tank	5
Volume control tank	5
Letdown flow rate to volume control tank	88 gpm
Time to isolate letdown flow	5 minutes
Letdown demineralizer decontamination factor	10

Table 9
Meteorological Data
used for
Radiological consequence Analyses

Exclusion Area Boundary [(1)]

Time (hr)	X/Q (sec/m^3)
0-2	2.232E-4

Low Population Zone Distance [(1)]

Time (hr)	X/Q (sec/m^3)
0 to 2	3.977E-5
2 to 24	4.100E-6
24 to 48	2.427E-6
48 to 720	4.473E-7

Control Room [(2)]

Time (hr)	X/Q (sec/m^3)
0 to 8	2.93E-3
8 to 24	1.73E-3
24 to 96	6.74E-4
96 to 720	1.93E-4

[(1)] Original licensing basis in Kewaunee USAR, Table 14.3-8
[(2)] Updated Control Room Habitability Evaluation Report, Table 5, based on “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19,” K. G. Murphy and K. M. Campe, 13th Air Cleaning Conference, August 1974.

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