October 27, 1998

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner, TSB
PD3- Reading	EGA1
ACRS	OGC
GGrant, RIII	CCarpenter
THarris (TLH3, SE only)	

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1499)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 8, 1998, as revised by letter dated August 27, 1998.

Generic Letter 95-05 supports the reduction of reactor coolant system specific activity limits as a means for accepting higher projected leakage rates from steam generator tubes while still meeting the limits for offsite dose (10 CFR 100) and control room dose (General Design Criterion 19). This amendment reduces Kewaunee's allowable RCS specific activity from 1.0 μCi/gram to 0.20 μCi/gram dose equivalent I-131. Corresponding bases pages are also being changed.

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

Sincerely,

Original signed by:

William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No.140 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\KEWAUNEE\PA153.AMD * See Previous Concurrence

OFFICE	PD31:PM	E	PD3 :LA	E	OGC		PD31:D	E
NAME	WLong W		EBarnhill EB		*		CACarpenter AC	
DATE	10/ /98		10/ /98		10/7/98		10/26/98	

OFFICIAL RECORD COPY

DF01

9811020136 981027
PDR ADOCK 05000305
P PDR

October 27, 1998

Distribution w/encls:

Docket File	GHill (2)
PUBLIC	WBeckner, TSB
PD3- Reading	EGA1
ACRS	OGC
GGrant, RIII	CCarpenter
THarris (TLH3, SE only)	

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1499)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 8, 1998, as revised by letter dated August 27, 1998.

Generic Letter 95-05 supports the reduction of reactor coolant system specific activity limits as a means for accepting higher projected leakage rates from steam generator tubes while still meeting the limits for offsite dose (10 CFR 100) and control room dose (General Design Criterion 19). This amendment reduces Kewaunee's allowable RCS specific activity from 1.0 μCi/gram to 0.20 μCi/gram dose equivalent I-131. Corresponding bases pages are also being changed.

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

Sincerely,

Original signed by:

William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 140 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\KEWAUNEE\PA153.AMD * See Previous Concurrence

OFFICE	PD31:PM	E	PD3 :LA	E	OGC		PD31:D	E
NAME	WLong WV		EBarnhill EB		*		CACarpenter AC	
DATE	10/27/98		10/27/98		10/7/98		10/26/98	

OFFICIAL RECORD COPY

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 27, 1998

Mr. M. L. Marchi
Site Vice President-Kewaunee Plant
Wisconsin Public Service Corporation
P.O. Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. MA1499)

Dear Mr. Marchi:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated April 8, 1998, as revised by letter dated August 27, 1998.

Generic Letter 95-05 supports the reduction of reactor coolant system specific activity limits as a means for accepting higher projected leakage rates from steam generator tubes while still meeting the limits for offsite dose (10 CFR 100) and control room dose (General Design Criterion 19). This amendment reduces Kewaunee's allowable RCS specific activity from 1.0 μCi/gram to 0.20 μCi/gram dose equivalent I-131. Corresponding bases pages are also being changed.

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

Sincerely,

William O. Long

William O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 140 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

M. L. Marchi
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

cc:

Foley & Lardner
ATTN: Bradley D. Jackson
One South Pinckney Street
P.O. Box 1497
Madison, WI 53701-1497

Chairman
Town of Carlton
Route 1
Kewaunee, WI 54216

Harold Reckelberg, Chairman
Kewaunee County Board
Kewaunee County Courthouse
Kewaunee, WI 54216

Attorney General
114 East, State Capitol
Madison, WI 53702

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Route #1, Box 999
Kewaunee, WI 54216-9511

Regional Administrator - Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4531

James D. Loock, Chief Engineer
Public Service Commission
of Wisconsin
P. O. Box 7854
Madison, WI 53707-7854

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated April 8, 1998, as revised by letter dated August 27, 1998, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

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

William O. Long

Wlilliam O. Long, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 27, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS 3.1-8	TS 3.1-8
TS B3.1-8	TS B3.1-8
TS B3.1-9	TS B3.1-9
TS B3.1-10	TS B3.1-10
TS B3.1-12	TS B3.1-12
TS B3.1-13	TS B3.1-13
TS B3.1-14	TS B3.1-14
Figure TS 3.1-3	Figure TS 3.1-3
TS B3.4-4	TS B3.4-4
Table TS 4.1-2, page 2 of 2	Table TS 4.1-2, page 2 of 2
TS B4.2-4	TS B4.2-4
TS B4.2-6	TS B4.2-6
TS B4.2-7	TS B4.2-7

c. Maximum Coolant Activity

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

   A. ≤ 0.20 μCi/gram DOSE EQUIVALENT I-131, and

   B. ≤ $\frac{91}{\bar{E}}$ $\frac{\mu Ci}{cc}$ gross radioactivity due to nuclides with half-lives > 30 minutes excluding tritium ($\bar{E}$ 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 > 0.20 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding the limit shown on Figure TS 3.1-3, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature of < 500°F within 6 hours.

   B. With the specific activity of the reactor coolant > $\frac{91}{\bar{E}}$ $\frac{\mu Ci}{cc}$ of gross radioactivity, be in at least INTERMEDIATE SHUTDOWN with an average coolant temperature < 500°F within 6 hours.

   C. With the specific activity of the reactor coolant > 0.20 μCi/gram DOSE EQUIVALENT I-131 *or* > $\frac{91}{\bar{E}}$ $\frac{\mu Ci}{cc}$ perform the sample and analysis requirements of Table TS 4.1-2, item 1.f, once every 4 hours until restored to within its limits.

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

If one train of RHR suction piping to RHR 33-1 is isolated, the valves and valve breakers in the other train shall be verified open, and the isolated flowpath must be restored within 5 days. If the isolated flowpath cannot be restored within 5 days, the RCS must be depressurized and vented through at least a 6.4 square inch vent within an additional 8 hours.

If both trains of RHR suction are isolated or valve RHR 33-1 is inoperable, the system can still be considered operable if an alternate vent path is provided which has the same or greater effective flow cross section as the LTOP safety valve (≥ 6.4 square inches). If vent path is provided by physical openings in the RCS pressure boundary (e.g., removal of pressurizer safety valves or steam generator manways), the vent path is considered secured in the open position.

Maximum Coolant Activity (TS 3.1.c)

The limit on gross specific activity is based on the evaluation of the consequences of a postulated rupture of a steam generator tube when the maximum activity in the reactor coolant is at the allowable limit. The potential release of activity to the atmosphere has been evaluated to insure that the public is protected.

Rupture of a steam generator tube would allow reactor coolant activity to enter the secondary system. The major portion of this activity is noble gases[18] which would be released to the atmosphere from the air ejector or a relief valve. Activity could continue to be released until the operator could reduce the Reactor Coolant System pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single tube, followed by isolation of the faulty steam generator by the operator within one-half hour after the event. During this period, 120,000 lbs. of reactor coolant are discharged into the steam generator.[18]

The limiting offsite dose is the whole-body dose resulting from immersion in the cloud containing the released activity. Radiation would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, for purposes of analysis, the simple model of a semi-infinite cloud, which gives an upper limit to the potential gamma dose, has been used. The semi-infinite cloud model is applicable to the beta dose because of the short range of beta radiation in air. The effectiveness of clothing as shielding against beta radiation is neglected and therefore the analysis model also gives an upper limit to the potential beta dose.

[18] USAR Section 14.2.4

The combined gamma and beta dose from a semi-infinite cloud is given by:

$$Dose,\ rem = 1/2\ [\bar{E} \cdot A \cdot V \cdot \frac{X}{Q} \cdot (3.7 \times 10^{10})\ \ (1.33 \times 10^{-11})]$$

Where: $\bar{E}$ = average energy of betas and gammas per disintegration (Mev/dis)

A = primary coolant activity (Ci/m^3)

$\bar{E}A$ = 91 Mev Ci/dis m^3 (the maximum per this specification)

$\frac{X}{Q}$ = 2.9 x 10^{-4} sec/m^3, the 0-2 hr. dispersion coefficient at the site boundary prescribed by the Commission

V = 77 m^3, which corresponds to a reactor coolant liquid mass of 120,000 lbs.

The resultant dose is < 0.5 rem at the site boundary.

Reactor coolant specific activity is further limited to ≤ 0.20 μCi/gram DOSE EQUIVALENT I-131 to ensure that offsite thyroid dose does not exceed 10 CFR 100 guidelines and that the control room thyroid dose does not exceed GDC-19. To ensure the allowable doses are not exceeded, an evaluation was performed to determine the maximum allowable primary to secondary leak rate which could exist during a steam line break event. This analysis is described in the Basis for TS 3.4.d on secondary activity limits.

The action statement permitting power operation to continue for limited time periods with reactor coolant specific activity > 0.20 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown in Figure TS 3.1-3, accommodates the possible iodine spiking phenomenon which may occur following changes in thermal power.

Reducing average coolant 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.

Leakage of Reactor Coolant (TS 3.1.d)[19]

TS (TS 3.1.d.1)

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System. Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 1 gpm unidentified leakage would not exceed the limits of 10 CFR Part 20. This is shown as follows:

If the reactor coolant activity is $91/\bar{E}$ μCi/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 1 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling System vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 2.0 \times 10^{-6}$ sec/m^3, is 0.09 rem/yr, compared with the 10 CFR Part 20 limits of 0.1 rem/yr.

With the limiting reactor coolant activity and assuming initiation of a 1 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room. Operators would then investigate the source of the leak and take actions necessary to isolate it. Should the leak result in a continuous discharge to the atmosphere via the component cooling surge tank and waste holdup tank, the resultant dose rate at the site boundary would be 0.09 rem/yr as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 1 gpm for an unidentified source of leakage is sufficiently above the minimum detectable leak rate to provide a reliable indication of leakage, and is well below the capacity of one charging pump (60 gpm).

Twelve (12) hours of operation before placing the reactor in the HOT SHUTDOWN condition are required to provide adequate time for determining whether the leak is into the containment or into one of the closed systems and to identify the leakage source.

TS 3.1.d.2

The 150 gpd leakage limit through any one steam generator is specified to ensure tube integrity is maintained in the event of a main steam line break or under loss-of-coolant accident conditions. This reduced operational leakage rate is applicable in conjunction with the tube support plate voltage-based plugging criteria as specified in TS 4.2.b.5.

---

(19) USAR Sections 6.5, 11.2.3, 14.2.4

Maximum Reactor Coolant Oxygen, Chloride and Fluoride Concentration (TS 3.1.e)

By maintaining the oxygen, chloride and fluoride concentrations in the reactor coolant below the limits as specified in TS 3.1.e.1 and TS 3.1.e.4, the integrity of the Reactor Coolant System is assured under all operating conditions.[20]

If these limits are exceeded, measures can be taken to correct the condition, e.g., replacement of ion exchange resin or adjustment of the hydrogen concentration in the volume control tank.[21] Because of the time-dependent nature of any adverse effects arising from oxygen, chloride, and fluoride concentration in excess of the limits, it is unnecessary to shut down immediately since the condition can be corrected. Thus, the time periods for corrective action to restore concentrations within the limits have been established. If the corrective action has not been effective at the end of the time period, reactor cooldown will be initiated and corrective action will continue.

The effects of contaminants in the reactor coolant are temperature dependent. The reactor may be restarted and operation resumed if the maximum concentration of any of the contaminants did not exceed the permitted transient values; otherwise a safety review by the Plant Operations Review Committee is required before startup.

Minimum Conditions for Criticality (TS 3.1.f)

During the early part of the fuel cycle, the moderator temperature coefficient may be calculated to be positive at ≤ 60% RATED POWER. The moderator coefficient will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is greatest. Later in the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients either will be less positive or will be negative.[22][23]

The requirement that the reactor is not to be made critical except as specified in TS 3.1.f.1 provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil-ductility temperature range. Heatup to this temperature will be accomplished by operating the reactor coolant pumps and by the pressurizer heaters.

---

[20] USAR Section 4.2

[21] USAR Section 9.2

[22] USAR Table 3.2-1

[23] USAR Figure 3.2-8

The shutdown margin specified in TS 3.10 precludes the possibility of accidental criticality as a result of an increase in moderator temperature or a decrease in coolant pressure.[22]

The requirement that the pressurizer is partly voided when the reactor is < 1% subcritical assures that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that the reactor is not to be made critical when the moderator coefficient is > 5.0 pcm/°F has been imposed to prevent any unexpected power excursion during normal operation, as a result of either an increase in moderator temperature or a decrease in coolant pressure. The moderator temperature coefficient limits are required to maintain plant operation within the assumptions contained in the USAR analyses. Having an initial moderator temperature coefficient no greater than 5.0 pcm/°F provides reasonable assurance that the moderator temperature coefficient will be negative at 60% rated thermal power. The moderator temperature coefficient requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[24] and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction in moderator density.

Suitable physics measurements of moderator coefficients of reactivity will be made as part of the startup testing program to verify analytical predictions.

Analysis has shown that maintaining the moderator temperature coefficient at criticality ≤ 5.0 pcm/°F will ensure that a negative coefficient will exist at 60% power. Current safety analysis supports operating up to 60% power with a moderator temperature coefficient ≤ 5.0 pcm/°F. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

[24] USAR Figure 3.2-9

The calculated hot full power (HFP) moderator temperature coefficient will be more negative than -8.0 pcm/°F for at least 95% of a cycle's time at HFP to ensure the limitations associated with and Anticipated Transient Without Scram (ATWS) event are not exceeded. NRC approved methods[25][26] will be used to determine the lowest expected HFP moderator temperature coefficient for the 5% of HFP cycle time with the highest boron concentration. The cycle time at HFP is the maximum number of days that the cycle could be at HFP based on the design calculation of cycle length. The cycle time at HFP can also be expressed in terms of burnup by converting the maximum number of days at full power to an equivalent burnup. If this HFP moderator temperature coefficient is more negative than -8.0 pcm/°F, then the ATWS design limit will be met for 95% of the cycle's time at HFP. If this HFP moderator temperature coefficient design limit is still not met after excluding the 5% of the cycle burnup with the highest boron concentration, then the core loading must be revised.

The results of this design limit consideration will be reported in the Reload Safety Evaluation Report.

In the event that the limits of TS 3.1.f.3 are not met, administrative rod withdrawal limits shall be developed to prevent further increases in temperature with a moderator temperature coefficient that is outside analyzed conditions. In this case, the calculated HFP moderator temperature coefficient will be made less negative by the same amount the hot zero power moderator temperature coefficient exceeded the limit in TS 3.1.f.3. This will be accomplished by developing and implementing administrative control rod withdrawal limits to achieve a moderator temperature coefficient within the limits for HFP moderator temperature coefficient.

Due to the control rod insertion limits of TS 3.10.d and potentially developed control rod withdrawal limits, it is possible to have a band for control rod location at a given power level. The withdrawal limits are not required if TS 3.1.f.3 is satisfied or if the reactor is subcritical.

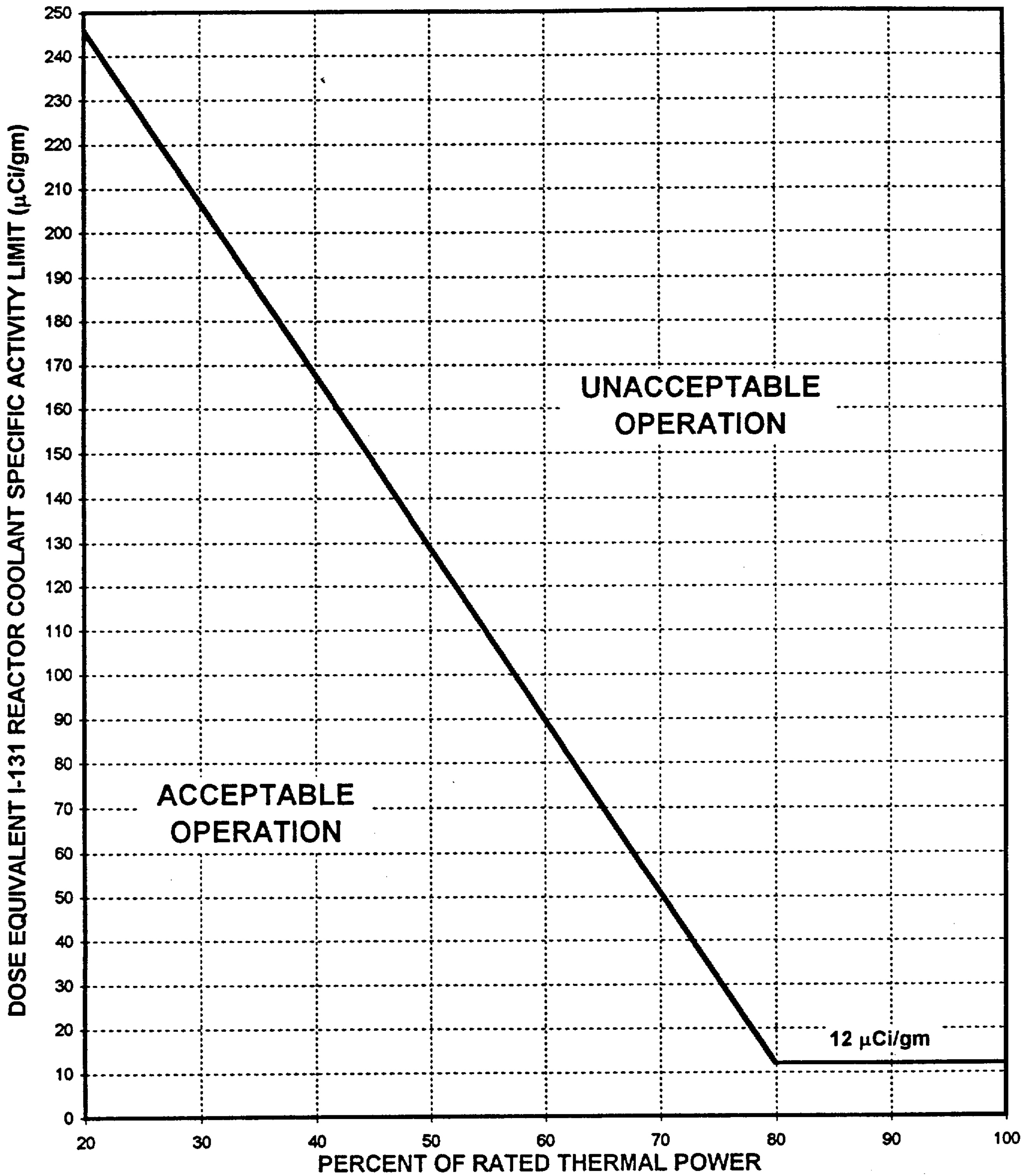
If after 24 hours, withdrawal limits sufficient to restore the moderator temperature coefficient to within the limits of TS 3.1.f.3 are not developed, the plant shall be taken to HOT STANDBY until the moderator temperature coefficient is within the limits of TS 3.1.f. The reactor is allowed to return to criticality whenever TS 3.1.f is satisfied.

---

[25] "NRC Safety Evaluation Report for Qualification of Reactor Physics, Methods for Application to Kewaunee," dated October 22, 1979.

[26] "NRC Safety Evaluation Report for the Reload Safety Evaluation Methods for Application to Kewaunee," dated April 11, 1988.

FIGURE TS 3.1-3



DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY > 0.20 μCi/GRAM DOSE EQUIVALENT I-131

Amendment No. ~~100~~,140

The cross-connect valves (AFW-10A and AFW-10B) are normally maintained in the open position. This provides an added degree of redundancy above what is required for all accidents except for a MSLB. During a MSLB, one of the cross-connect valves will have to be repositioned regardless if the valves are normally opened or closed. Therefore, the position of the cross-connect valves does not affect the performance of the turbine-driven AFW train. However, performance of the train is dependent on the ability of the valves to reposition. Although analyses have demonstrated that operation with the cross-connect valves closed is acceptable, the TS restrict operation with the valves closed to <15% of RATED POWER. At ≥ 15% RATED POWER, closure of the cross-connect valves renders the TDAFW train inoperable.

An AFW train is defined as the AFW system piping, valves and pumps directly associated with providing AFW from the AFW pumps to the steam generators. The action with three trains inoperable is to maintain the plant in an operating condition in which the AFW System is not needed for heat removal. When one train is restored, then the LIMITING CONDITIONS FOR OPERATION specified in TS 3.4.b.2 are applied. Should the plant shutdown be initiated with no AFW trains available, there would be no feedwater to the steam generators to cool the plant to 350°F when the RHR System could be placed into operation.

It is acceptable to exceed 350°F with an inoperable turbine-driven AFW train. However, OPERABILITY of the train must be demonstrated within 72 hours after exceeding 350°F or a plant shutdown must be initiated.

Condensate Storage Tank (CST)(TS 3.4.c)

The specified minimum water supply in the condensate storage tanks (CST) is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

The shutdown sequence of TS 3.4.c.3 allows for a safe and orderly shutdown of the reactor plant if the specified limits cannot be met.

Secondary Activity Limits (TS 3.4.d)

An evaluation was performed to determine the maximum permissible steam generator primary-to-secondary leak rate during a steam line break event. The evaluation considered both a preaccident and accident initiated iodine spike for offsite dose and control room operator dose. The results of the evaluation show that the control room operator dose with an accident initiated spike yields the limiting leak rate. This evaluation was based on a 30 REM thyroid dose and initial primary and secondary coolant iodine activity levels of 0.20 μCi/gm and 0.1 μCi/cc DOSE EQUIVALENT I-131 respectively. A leak rate of 9.0 gpm was determined to be the upper limit for allowable primary-to-secondary leakage in the steam generator faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm (150 gpd per TS 3.1.d.2), the standard operating leakage limit applied for the tube support plate voltage-based plugging criteria specified in TS 4.2.b.5.

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

SAMPLING TESTS	TEST	FREQUENCY
3. Refueling Water Storage Tank Water Sample[7]	Boron Concentration	Monthly[8]
4. Deleted		
5. Accumulator	Boron Concentration	Monthly
6. Spent Fuel Pool	Boron Concentration	Monthly[9]
7. Secondary Coolant	a. Gross Beta or Gamma Activity b. Iodine Concentration	Weekly Weekly when gross beta or gamma activity $\geq 0.1\ \mu Ci/cc$

[7] A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during REFUELING outages.

[8] And after adjusting tank contents.

[9] Sample will be taken monthly when fuel is in the pool.

The pressure boundary for HEJ sleeves is shown on Figure TS 4.2-1. The pressure boundary used to disposition parent tube indications (PTIs) detected in the upper joint of HEJ sleeved tubes is discussed in WCAP-15050.[5]* The pressure boundary described in the WCAP will allow PTIs located in the upper joint to remain in service if there is a minimum non-degraded (i.e., no detectable degradation in the parent tube) hardroll length of 0.92 inch (plus an allowance for NDE uncertainty) as measured from the bottom of the hardroll upper transition. The minimum hardroll engagement length is derived from structural and leakage testing. During field application, the PTI is located in reference to the bottom of the hardroll upper transition to ensure the minimum length of non-degraded hardroll exits. The inspection is performed using eddy current techniques capable of profiling and flaw detection as described in "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications."[6] The NDE uncertainty for this criterion is a function of the eddy current probe and technique used. The uncertainty has been calculated to be 0.023 inch. However, for field application, an eddy current uncertainty of 0.03 inch will be applied to the minimum hardroll engagement length of 0.92 inch.

Leakage testing performed for the HEJ pressure boundary showed that leak rates for normal operating and steam line break (SLB) are comparable. However, statistical analysis shows that for a 99 percent confidence level the ratio of leak rate at SLB to normal operating is 9.3.[5] To bound SLB leak rate, the assumption is made that SLB leak rate is one order of magnitude greater than normal operating leak rate. The normal operating primary-to-secondary leakage limit is 0.104 gpm (150 gpd per TS 3.1.d.2). Therefore, the maximum primary-to-secondary leak rate during a SLB is assumed to be approximately 1 gpm (9.3 x 0.104 gpm). The 1 gpm will be the assigned leakage encompassing the HEJs left in service using the length criterion described in the paragraph above. Steam line break leakage in the faulted loop from all sources must be calculated to be less than or equal to the maximum allowable leakage described in the Basis for TS 3.4.d. Maintenance of the maximum allowable leak rate limit ensures offsite doses will remain within a small fraction of the 10 CFR Part 100 guidelines and ensures control room doses will not exceed GDC-19 during a SLB.

---

[5] WCAP-15050, "HEJ Sleeved Tube Length Based Degradation Acceptance Criterion," May 1998.

*The pressure boundary described by WCAP-15050 is applicable for operating cycle 23 only.

[6] "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications," Attachment 5 to Letter to Document Control Desk from C.R. Steinhardt dated May 14, 1998.

The tube support plate sleeve is 12" long and spans the degraded area of the tube adjacent to the support plate intersection. The tube support plate sleeve is hydraulically expanded and laser welded at each end. The pressure boundary portion of the tube support plate sleeve is the weld and the sleeve section between the welds. Tubesheet sleeves extend from the tube end to above the top of the tubesheet. Standard and bowed or peripheral tubesheet sleeves can be installed. The upper or free span joint is hydraulically expanded and laser welded. The lower joint is hydraulically expanded and roll expanded. Standard tubesheet sleeves extend from 27" to 36" in length while bowed tubesheet sleeves extend from 30" to 36" in length. The pressure boundary portion of the tubesheet sleeve is the weld and below, down to the tubesheet primary face.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

<u>Technical Specification 4.2.b.5</u>

The repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 and GDC-19 limits are not exceeded.

The voltage-based repair limits of TS 4.2.b.5 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit, V_{SL}, is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

Where V_{GR} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

Technical Specification 4.2.b.6

Tubes with indications of degradation in either the original factory roll expansion in the tubesheet or the unexpanded portion of tube within the tubesheet may be dispositioned for continued service or repaired through application of the F* or EF* criteria. The F* and EF* criteria are described in WCAP-14677.[11] The F* and EF* criteria are established using guidance consistent with RG 1.121. Neither the F* or EF* criteria will significantly contribute to offsite dose following a postulated main steam line break such that contributions from these sources need to be included in offsite dose analyses. Inherent to these criteria is the ability to perform an additional roll expansion of the tube, either as an extension of the original factory roll expansion, in which case F* criteria applies, or in the area starting approximately 4 inches below the top of the tubesheet, in which case EF* criterion apply. The additional roll expansion procedure can be applied over existing degradation, provided the F* or EF* requirements for non-degraded roll expansion lengths of 1.11 inches (plus an allowance for NDE uncertainty) and 1.51 inches (plus an allowance for NDE uncertainty), respectively, are satisfied. The NDE uncertainty applied to the F* and EF* distance is a function of the eddy current probe and technique used. Current state-of-the art inspection technology will be used with implementation of the F* and EF* criteria. The uncertainty in such inspections has been shown to be as small as 0.06 inches, however, for field application, an eddy current uncertainty of 0.20 inches will be applied. Any and all indications of degradation existing below the F* or EF* distance is acceptable for continued service.

---

[11]WCAP 14677, Revision 1, F* and Elevated F* Tube Alternate Repair Criteria for Tubes With Degradation Within the Tubesheet Region of the Kewaunee Steam Generators, May 1998 (Proprietary).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

The Kewaunee Nuclear Power Plant has been approved to use a voltage-based repair criteria for the steam generator tube support plate intersections (ref: Amendment 126 issued September 11, 1996). The use of voltage-based repair criteria is discussed in Generic Letter 95-05. One of the features of Generic Letter 95-05 is the use of more restrictive reactor coolant activity limits to account for the possibility of increased leakage rates during faulted conditions. In its license amendment application dated April 8, 1998, the licensee requested that the Technical Specifications (TS) be changed to reduce the reactor coolant system (RCS) dose equivalent iodine-131(^{131}I) limits from 1.0 μCi/g to 0.35 μCi/g for the 48-hour limit and from 60 μCi/g to 21 μCi/g for the maximum instantaneous limit. The allowable activity level of dose equivalent ^{131}I in the secondary coolant was assumed to be equal to the TS limit of 0.1 μCi/g. The licensee also requested that Kewaunee be approved to operate based upon a 12.85 gpm (at room temperature and pressure) primary to secondary leak initiated by an accident in the faulted steam generator and the TS allowable value for primary to secondary system leakage from the intact steam generator of 150 gpd. As part of this amendment request, the licensee performed an assessment of the radiological dose consequences of a main steam line break (MSLB) accident. The licensee found the radiological dose consequences of incorporating these proposed changes to be acceptable based on the NRC acceptance criteria for doses at the Exclusion Area Boundary (EAB), the Low-Population Zone (LPZ), and the control room.

2.0 EVALUATION

The staff reviewed the licensee's calculations and found that the licensee had assumed that the accident initiated iodine spike is terminated after 1.7 hours. The licensee stated that 1.7 hours is the time at which the iodine concentration in the RCS would reach 60 times the maximum equilibrium RCS TS concentration of 0.35 μCi/g of dose equivalent ^{131}I. This is an incorrect assumption since the limit of 60 times the maximum equilibrium RCS TS concentration does not apply for the accident initiated case. In performing an MSLB accident analysis, the staff assumes that the accident initiated spike lasts for the full 8-hour duration of a MSLB accident, unless it can be shown that the iodine in the fuel gap is exausted prior to this 8-hour period. For the case of Kewaunee, the staff calculated that the iodine in the fuel gap would not be exhausted prior to 8 hours. Using an accident initiated spike with a duration of 8 hours, the staff calculated an unacceptably high dose to the control room operator for the accident initiated spike case.

In a letter to the staff dated August 27, 1998, the licensee stated that they would need to revise the iodine concentrations and leakage values proposed in their earlier April 8, 1998, submittal in order to achieve acceptable dose consequences when using an accident initiated spike of 8 hours duration. The licensee proposed to reduce the RCS dose equivalent ^{131}I limits from the proposed 0.35 μCi/g to 0.20 μCi/g for the 48-hour limit (from 21 μCi/g to 12 μCi/g for the maximum instantaneous limit) and decrease the primary to secondary leakage from the faulted SG from the proposed 12.85 gpm to 9 gpm.

The staff performed confirmatory calculations to check the acceptability of the licensee's revised numbers. The staff calculated the doses resulting from an MSLB accident using the methodology associated with Standard Review Plan 15.1.5, Appendix A. The staff performed two separate assessments. One was based upon a pre-existing iodine spike activity level of 12 μCi/g of dose equivalent ^{131}I in the primary coolant and the other was based upon an accident initiated iodine spike. For the accident initiated spike case, the staff assumed that the primary coolant activity level was 0.2 μCi/g of dose equivalent ^{131}I. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the normal release rate to maintain an activity level of 0.2 μCi/g of dose equivalent ^{131}I in the primary coolant. For these two cases, the staff calculated the thyroid doses for individuals located at the EAB and at the LPZ. The staff also calculated the thyroid dose to the control room operator. The parameters which were utilized in the staff's assessment are presented in Table 1. The EAB, LPZ, and control room doses calculated by the staff are presented in Table 2.

The staff's calculations confirmed that the doses from a postulated MSLB accident meet the acceptance criteria and that the licensee's calculations are acceptable. The results of both the licensee's and staff's calculations showed that the thyroid doses at the EAB and LPZ would be less than the guidelines established by SRP 15.1.5, Appendix A of NUREG-0800 (acceptance criterion of 300 rem thyroid dose at the EAB and LPZ for the pre-existing spike case and 30 rem thyroid dose at the EAB and LPZ for the accident initiated spike case). The calculated thyroid dose to the control room operator was less than the guidelines of SRP 6.4 of NUREG-0800 (acceptance criterion of 30 rem thyroid to the control room operator). On this basis, the staff approves the requested reduction in the allowable reactor coolant system dose equivalent ^{131}I activity from 1.0 to 0.2 μCi/g along with the proposed allowable maximum primary to secondary system coolant leakage of 9.0 gpm (at room temperature and pressure) initiated in the faulted SG by a MSLB accident.

3.0 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.

4.0 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 or changes a surveillance requirement. 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 (63 FR 49137). 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.

5.0 CONCLUSION

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

Principal Contributor: C. Hinson

Date: October 27, 1998

TABLE 1
INPUT PARAMETERS FOR KEWAUNEE
EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary Coolant Concentration of 12 μCi/g of Dose Equivalent ^{131}I

   <u>Pre-existing Spike Value (μCi/g)</u>

   ^{131}I = 9.22
   ^{132}I = 3.67
   ^{133}I = 15.14
   ^{134}I = 2.33
   ^{135}I = 8.33

2. Data on Primary Coolant and Secondary Coolant

Primary Coolant Volume (ft^3)	6,236
Primary Coolant Temperature (°F)	578
Secondary Coolant Liquid Volume (ft^3/SG)	1,920
Secondary Coolant Steam Volume (ft^3/SG)	3,838
Secondary Coolant Steam Temperature (°F)	510.8
Feedwater Temperature (°F)	427.3

3. TS Limits for DE ^{131}I in the Primary and Secondary Coolant

Maximum Instantaneous DE ^{131}I Concentration (μCi/g)	12.0
Primary Coolant DE ^{131}I Concentration (μCi/g)	0.2
Secondary Coolant DE ^{131}I Concentration (μCi/g)	0.1

4. TS Value for the Primary to Secondary Leak Rate

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, both SGs (gpd)	300

5. Maximum Primary to Secondary Leak Rate to the Faulted and Intact SGs

Faulted SG (gpm)	9.0
Intact SG (gpm/SG)	0.1

6. Iodine Partition Factor (for steaming of SG water)

Faulted SG	1.0
Intact SG	0.01

7. Steam Released to the Environment (lbs)

Intact SG (0 - 2 hours)	290,000
Intact SG (2 - 8 hours)	433,254
Faulted SG (0 - 15 minutes)	99,300

8. Letdown Flow Rate (gpm) 40

9. Release Rate for 0.2 μCi/g of Dose Equivalent ^{131}I

	Release Rate (Ci/hr)	500X Release Rate (Ci/hr)
^{131}I =	1.04	523
^{132}I =	2.69	1,346
^{133}I =	2.66	1,331
^{134}I =	4.03	2,017
^{135}I =	2.7	1,349

10. Atmospheric Dispersion Factors

	sec/m^3
EAB (0-2 hours)	* 2.9×10^{-4}
LPZ (0-8 hours)	* 5.2×10^{-5}
Control Room (0-8 hours)	2.93×10^{-3}

11. Control Room Parameters

Filter Efficiency(%)	
Air recirculation filter	90
Volume (ft^3)	127,600
Makeup flow (cfm)	0
Filtered Recirculation Flow (cfm)	2,500
Unfiltered Inleakage (cfm)	200
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

* NRC staff calculated values

Table 2 - THYROID DOSES FROM KEWAUNEE MAIN STEAM LINE BREAK ACCIDENT (REM) (VALUES CALCULATED BY NRC STAFF)

	DOSE	
LOCATION	**Pre-Existing Spike**	**Accident-Initiated Spike****
EAB	3.75*	2.39
LPZ	2.52*	5.48
Control Room **	12.0	26.8

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid