ATTACHMENT 1

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 27, 1998

Proposed Amendment 153a

Introduction
Description of Proposed Changes
Safety Evaluation
Significant Hazards Determination
Environmental Considerations

INTRODUCTION

Wisconsin Public Service Corporation (WPSC) is submitting this proposed Technical Specification (TS) amendment request to reduce the reactor coolant system (RCS) specific activity limits in accordance with Generic Letter (GL) 95-05 (reference 1). This change will reduce allowable RCS specific activity from 1.0 μCi/gram to 0.20 μCi/gram dose equivalent I-131. Technical specification amendment 126 (reference 2), approved September 11, 1996, incorporated the voltage-based steam generator (SG) repair criteria for the tube support plate elevations in accordance with GL 95-05. The GL supports the reduction of RCS specific activity limits as a means for accepting higher projected leak rates for SG tubes while still meeting the limits for offsite dose (10 CFR 100) and control room dose (GDC-19). Technical specification amendment 126 did not implement the option of reduced specific activity. Since 1990, in support of various SG licensing actions, several iterations of the maximum allowable leak rate calculation have been performed. The most recent of the leak rate calculations did not address control room operator dose; therefore, WPSC has elected to reperform the leak rate and dose calculation following the guidance in GL 95-05. During this final iteration it was decided to reduce allowable reactor coolant specific activity from 1.0 μCi/gram to 0.20 μCi/gram dose equivalent I-131. By lowering RCS specific activity, a higher allowable projected leak rate can be accepted.

Along with these changes, the table which specifies the minimum frequencies for sampling the secondary coolant will be changed. In addition to incorporating the voltage-based repair, Technical specification amendment 126 also reduced secondary coolant activity from 1.0 μCi/cc to 0.1 μCi/cc. However, due to an oversight, the table mentioned above was not revised to reflect that change. Additionally, bases pages TS B3.1-8 through TS B3.1-14 are being updated to correct a misnumbered footnote. This is an administrative change.

DESCRIPTION OF THE PROPOSED CHANGE

This proposed amendment (PA) modifies the following Kewaunee Nuclear Power Plant (KNPP) TS:

1. TS 3.1.c.1.A, 3.1.c.2.A, and 3.1.c.2.C are being revised to lower reactor coolant specific activity from 1.0 μCi/gram dose equivalent I-131 to 0.20 μCi/gram dose equivalent I-131. This change is based on a calculation performed by Westinghouse to determine the maximum permissible primary to secondary leak rate during a postulated main steam line break (MSLB) event. The calculation is based on limiting offsite (i.e., site boundary and low population zone doses) and control room doses to 30 rem to the thyroid. The proposed specific activity level and the supporting calculation follow guidance suggested by GL 95-05.

2. Figure TS 3.1-3 is being revised to reflect the new reduced limit on reactor coolant specific activity. The curve on the graph is being lowered by a factor of 0.80 as applied to specific activity limits of Westinghouse standard technical specifications (STS). This change results in a full power specific activity limit of 12 μCi/gram dose equivalent I-131 as compared with the STS limit of 60 μCi/gram dose equivalent I-131. The new curve parallels the STS curve. The change is in accordance with the GL 95-05 guidance and is supported by the calculation.

3. Table TS 4.1-2 is being revised to reduce the secondary coolant activity for sampling test 7.b from 1.0 μCi/cc to 0.1 μCi/cc. It appears that when the secondary coolant activity limit (TS 3.4.d) was reduced in TS amendment 126, this table was not revised. The correct secondary coolant activity, 0.1 μCi/cc, was used in the calculation performed by Westinghouse for maximum permissible primary to secondary leak rate.

Concurrent with these changes, the appropriate bases pages have been revised and are being submitted for your information. Basis pages for TS 3.1.c are being updated to reflect the new RCS specific coolant activity. The basis page for TS 3.4.d, "Secondary Activity Limits," is being revised to accurately reflect the new analysis performed by Westinghouse for allowable leak rate. Bases pages for TS 4.2.b.4 and 4.2.b.5 for SG repair are also being updated to reflect the new analysis. Additionally, bases pages TS B3.1-8 through TS B3.1-14 are being updated to correct a misnumbered footnote. This is an administrative change to the bases pages.

SAFETY EVALUATION

The proposed change in RCS specific activity is supported by a calculation performed by Westinghouse to determine the maximum permissible primary to secondary leak rate during a MSLB event. The evaluation considered doses at the site boundary, the low population zone and the control room. It also considered both a pre-accident and accident initiated iodine spike. The results of the evaluation show that the control room dose with an accident initiated iodine spike yields the limiting leak rate. The evaluation was based on a 30 rem thyroid dose and initial primary and secondary coolant activity levels of 0.20 μCi/gram and 0.1 μCi/cc dose equivalent iodine-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 for the faulted loop. The steam generator in the intact loop was assumed to leak at a rate of 0.1 gpm (150 gpd), the standard operating leakage limit applied for voltage-based repair criteria. A summary of key parameters for the evaluation is included in Attachment 4.

The reduction of primary coolant activity is also supported by GL 95-05 as a means for accepting higher projected leakage rates while still meeting the applicable limits of 10 CFR 100 and GDC 19 criteria using licensing basis assumptions. The above described calculation ensures the 10 CFR 100 and GDC 19 criteria are met for this change.

SIGNIFICANT HAZARDS DETERMINATION

This proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1) Involve a significant increase in the probability or consequence of an accident previously evaluated.

   The change implements a more restrictive RCS activity limit. Specific RCS activity is an initial plant condition and, therefore, is not an accident initiator and can not cause the occurrence of or increase the probability of an accident. The change also lowers the curve of Figure TS 3.1-3 which restricts operation with high specific activity. The new value for specific activity is justified by the Westinghouse calculation which demonstrates acceptable offsite and control room doses following a MSLB with a maximum allowable primary to secondary leak rate. By lowering the RCS specific activity and maintaining leakage within the projected maximum allowable, 10 CFR 100 and GDC 19 criteria are satisfied. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any previously evaluated.

   The proposed change to the RCS specific activity limit will not significantly effect operation of the plant nor will it alter the configuration of the plant. There will be no additional challenges to the main steam system or the reactor coolant system pressure boundary and no new failure modes are introduced. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in the margin of safety.

   Reduction of the RCS specific activity limit allows an increase in the MSLB allowable primary to secondary leakage. The net effect is no reduction in the margin of safety provided by 10 CFR 100 and GDC 19 criteria. The maximum allowable leakage is the leakage limit for projected SG leakage following SG tube inspection and repair. Reducing specific activity to increase projected leak rate follows guidance given by GL 95-05 and effectively takes margin available in the specific activity limits and applies it to the projected SG leak rate. This has been determined to be an acceptable means for accepting higher projected leak rates while still meeting the applicable limits of 10 CFR 100 and GDC 19 criteria with respect to offsite and control room doses. Additionally, monitoring of the specific activity and compliance with the required actions remains unchanged. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

For consistency, the value of secondary coolant activity in Table TS 4.1.2 is being corrected from 1.0 μCi/gram to 0.1 μCi/gram. This is consistent with a previously submitted and approved amendment, therefore, no significant hazards exist for this change.

ENVIRONMENTAL CONSIDERATIONS

This proposed amendment involves a change to a requirement with respect to the use of a facility component located within the restricted area and a surveillance requirement. Wisconsin Public Service Corporation has determined that the proposed amendment involves no significant hazards considerations 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. Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no enviroumental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

ATTACHMENT 2

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 27, 1998

PROPOSED TS AMENDMENT NO. 153a

Strike Out TS Pages:

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

c. Maximum Coolant Activity

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

   A. $\leq$ ~~1.0~~0.20 μCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ 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 > ~~1~~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 > ~~1.0~~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.

<u>Maximum Coolant Activity (TS 3.1.c)</u>

~~This specification~~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~~[17]~~[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.~~[17]~~[18]

The limiting ~~off-site~~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.

~~[17]~~
[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)

$\frac{X}{Q}$ = 2.9×10^{-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 > ~~1~~0.20 μCi/gram~~s~~ 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)~~[18]~~[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.

~~[18]~~

[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.~~[19]~~[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.~~[20]~~[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.~~[21][22]~~[22][23]

~~[19]~~

[20]USAR Section 4.2

~~[20]~~

[21]USAR Section 9.2

~~[21]~~

[22]USAR Table 3.2-1

~~[22]~~

[23]USAR Figure 3.2-8

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.

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.~~(21)~~(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~~(20)~~(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 $\leq$ 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 $\leq$ 5.0 pcm/°F. At power levels greater than 60%, a negative moderator temperature coefficient must exist.

~~(23)~~

(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~~(24)(25)~~(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.

~~(24)~~

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

~~(25)~~

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

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.

<u>Condensate Storage Tank (CST)(TS 3.4.c)</u>

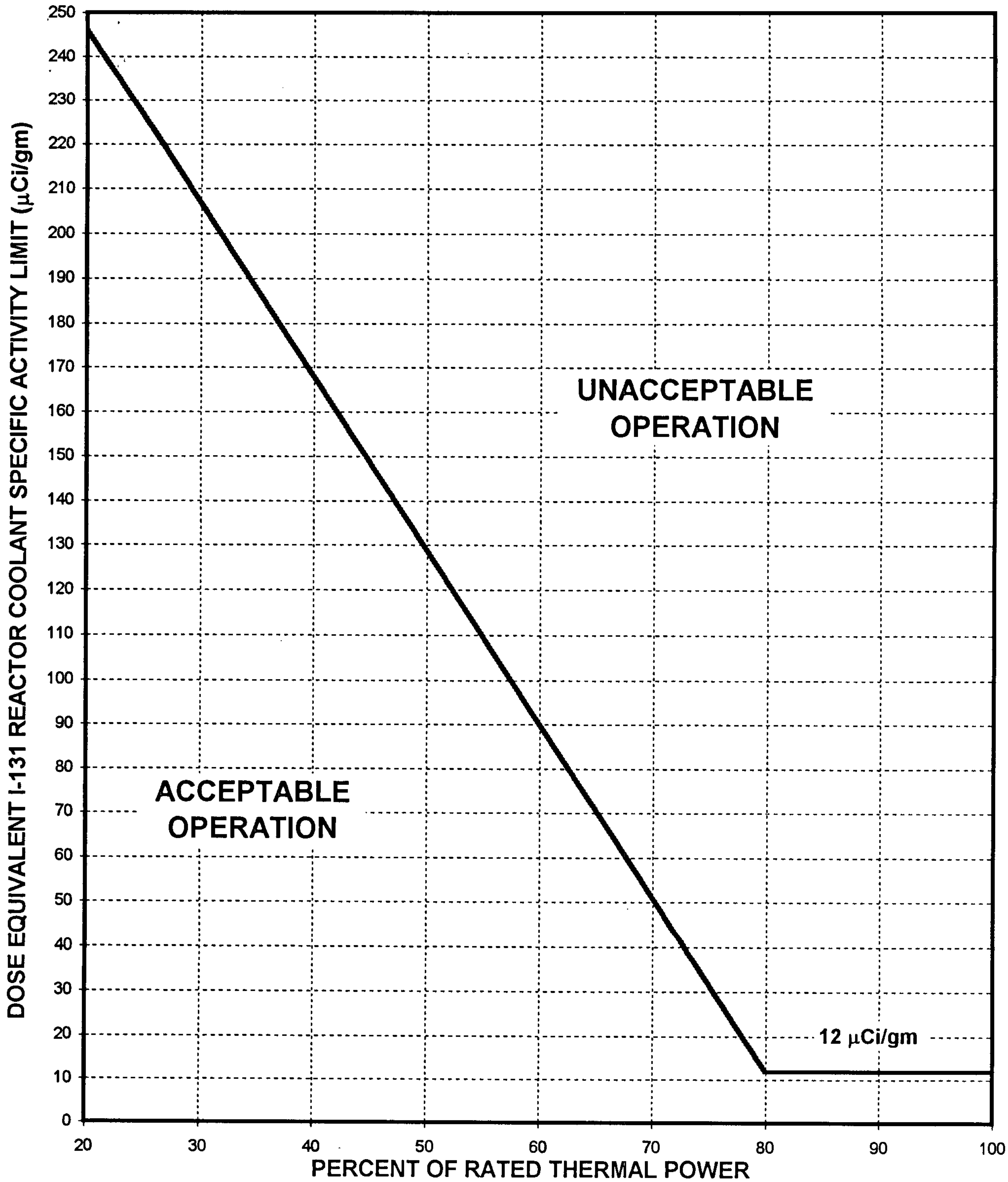
The specified minimum water supply in the condensate storage tanks (CST) is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

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.

<u>Secondary Activity Limits (TS 3.4.d)</u>

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 ~~at the site boundary~~ and initial primary and secondary coolant iodine activity levels of ~~1.0~~0.20 μCi/gm and 0.1 μCi/~~gm~~cc DOSE EQUIVALENT I-131 respectively. A leak rate of ~~34.0~~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.

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

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-14641.[5] The pressure boundary will allow PTIs located such that there is a minimum diameter change of 0.003 inch (plus an allowance for NDE uncertainty) between the peak diameter of the sleeve hardroll, and the diameter at the elevation of the PTI, to remain in service. The 0.003 inch interference lip is derived from structural and leakage testing. When inspecting and dispositioning the PTIs, the acceptance criteria will be adjusted to account for measurement uncertainties associated with the technique used to measure the relative change in ID sleeve diameters. During field application, the PTI elevation will be measured by comparing the diameter reported at the peak amplitude of the flaw, and the diameter at the center of the plus point coil's field, and using the more conservative of the two diameters to perform the ΔD determination. Application of the pressure boundary for HEJ sleeved tubes provides allowance for leakage in a faulted loop during a postulated steam line break (SLB) event. A SLB leakage of 0.025 gpm is assumed for each applicable indication. Steam line break leakage in the faulted loop from all sources must be calculated to be ~~< 34 gpm~~ less than or equal to the maximum allowable leakage described in the Basis for TS 3.4.d~~in the faulted loop~~. Maintenance of the ~~34 gpm~~maximum allowable leak rate limit ensures offsite doses will remain within a small fraction of the 10 CFR Part 100 guidelines ~~for~~and ensures control room doses will not exceed GDC-19 during a SLB.

Recent inspection information has indicated a potential for the parent tube behind the upper HEJ region to develop service induced degradation. For parent tube degradation within or below the upper HEJ hardroll lower transition, tube operability can be restored by fusing the sleeve and tube using a laser welding process effectively isolating the degradation below the weld. The laser weld repair is performed similar to the initial installation of laser welded sleeves. The laser repair weld for degraded parent tubes with installed HEJ sleeves has been shown to meet the weld qualification, stress and fatigue requirements of the ASME code. All laser weld repaired HEJ sleeved tubes will receive a post weld stress relief at the weld location and ultrasonic inspection to verify weld quality, in accordance with the process described in WCAP-14685, Revision 3[6] and WCAP-14685, Revision 2, Addendum 1.[7]

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[5]WCAP-14641, "HEJ Sleeved Tube Structural Integrity Criteria: Diameter Interference at PTIs," April 1996.

[6]WCAP-14685, Revision 3, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant," May 1997 (Proprietary).

[7]WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes," April 1997 (Proprietary).

Technical Specification 4.2.b.5

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.

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 ≥ ~~1.0~~0.1 μCi/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.

ATTACHMENT 3

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 27, 1998

PROPOSED TS AMENDMENT NO. 153a

Affected TS Pages:

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

c. Maximum Coolant Activity

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

   A. $\leq$ 0.20 μCi/gram DOSE EQUIVALENT I-131, and

   B. $\leq \frac{91}{\bar{E}} \frac{\mu Ci}{cc}$ 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-I 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)

$\frac{X}{Q}$ = 2.9×10^{-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 $91/\bar{E}\,\mu Ci/cc$ ($\bar{E}$ = 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.

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[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.

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[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.

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[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.

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[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.

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 $\geq$ 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.

<u>Condensate Storage Tank (CST)(TS 3.4.c)</u>

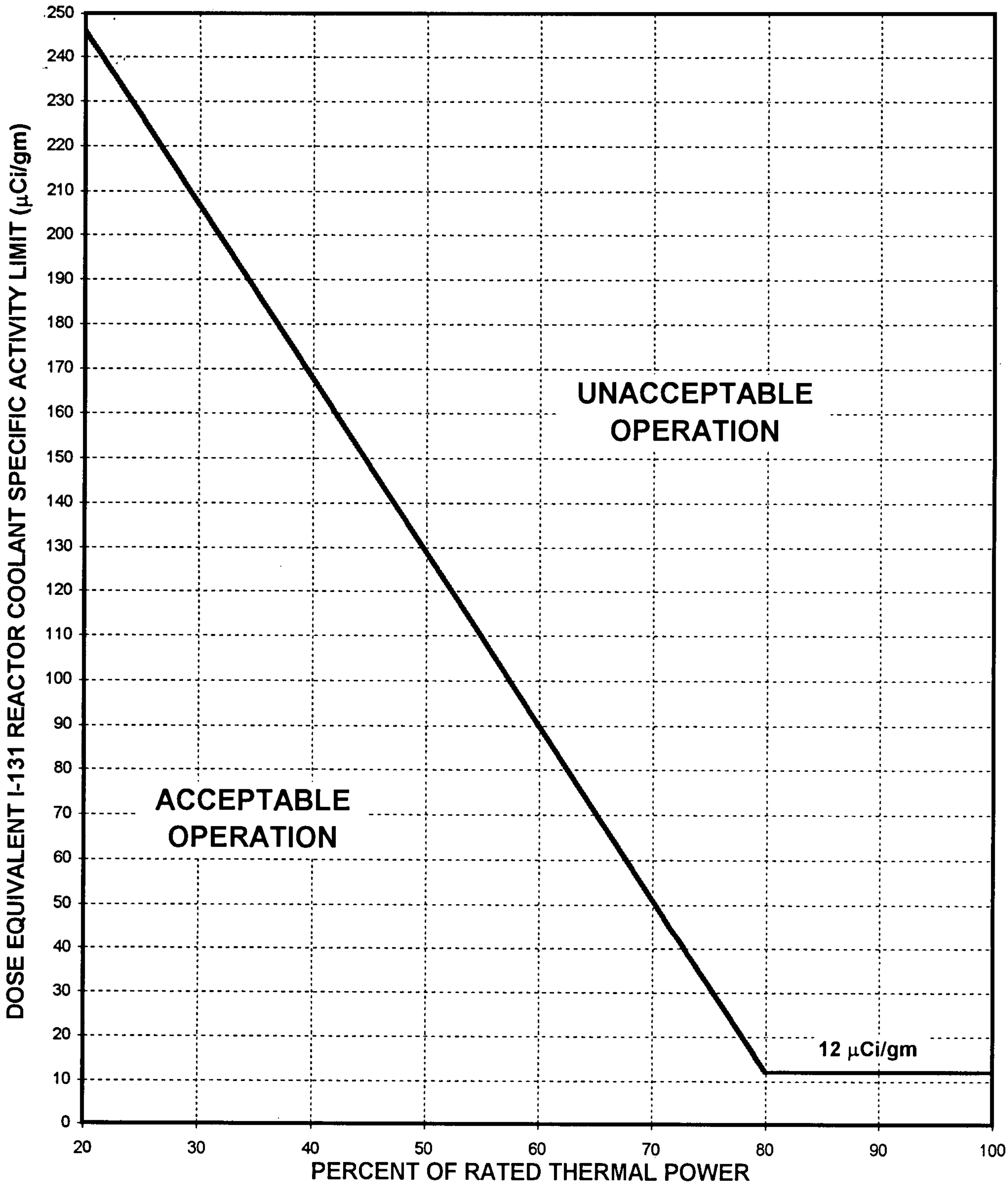
The specified minimum water supply in the condensate storage tanks (CST) is sufficient for 4 hours of decay heat removal. The 4 hours are based on the Kewaunee site specific station blackout (loss of all AC power) coping duration requirement.

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.

<u>Secondary Activity Limits (TS 3.4.d)</u>

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.

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

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-14641.[5] The pressure boundary will allow PTIs located such that there is a minimum diameter change of 0.003 inch (plus an allowance for NDE uncertainty) between the peak diameter of the sleeve hardroll, and the diameter at the elevation of the PTI, to remain in service. The 0.003 inch interference lip is derived from structural and leakage testing. When inspecting and dispositioning the PTIs, the acceptance criteria will be adjusted to account for measurement uncertainties associated with the technique used to measure the relative change in ID sleeve diameters. During field application, the PTI elevation will be measured by comparing the diameter reported at the peak amplitude of the flaw, and the diameter at the center of the plus point coil's field, and using the more conservative of the two diameters to perform the ΔD determination. Application of the pressure boundary for HEJ sleeved tubes provides allowance for leakage in a faulted loop during a postulated steam line break (SLB) event. A SLB leakage of 0.025 gpm is assumed for each applicable indication. 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.

Recent inspection information has indicated a potential for the parent tube behind the upper HEJ region to develop service induced degradation. For parent tube degradation within or below the upper HEJ hardroll lower transition, tube operability can be restored by fusing the sleeve and tube using a laser welding process effectively isolating the degradation below the weld. The laser weld repair is performed similar to the initial installation of laser welded sleeves. The laser repair weld for degraded parent tubes with installed HEJ sleeves has been shown to meet the weld qualification, stress and fatigue requirements of the ASME code. All laser weld repaired HEJ sleeved tubes will receive a post weld stress relief at the weld location and ultrasonic inspection to verify weld quality, in accordance with the process described in WCAP-14685, Revision 3[6] and WCAP-14685, Revision 2, Addendum 1.[7]

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[5] P-14641, "HEJ Sleeved Tube Structural Integrity Criteria: Diameter Interference at PTIs," April 1996.

[6] WCAP-14685, Revision 3, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant," May 1997 (Proprietary).

[7] WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes," April 1997 (Proprietary).

Technical Specification 4.2.b.5

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.

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 ≥ 0.1 μCi/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.

ATTACHMENT 4

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

August 27, 1998

PROPOSED TS AMENDMENT NO. 153a

Key Parameters - Report for Maximum Allowable
Primary to Secondary Leak Rate

Steamline Break Radiologieal Consequences

Introduction

For this analysis the complete severance of a main steamline outside containment is assumed to occur. The affected SG will rapidly depressurize and release radioiodines initially contained in the secondary coolant and primary coolant activity, transferred via SG tube leaks, directly to the outside atmosphere. A portion of the iodine activity initially contained in the intact SGs is released to atmosphere through either the atmospheric dump valves (ADV) or the safety valves (MSSVs). This analysis evaluated the maximum permissible primary to secondary leak rate which could exist in the faulted SG without exceeding the allowable dose rates at the site boundary, the low population zone or the control room. This section describes the assumptions and analyses performed to determine the amount of radioactivity released and the offsite and control room doses resulting from this release.

Input Parameters and Assumptions

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). For the pre-accident iodine spike it is assumed that a reactor transient has occurred prior to the SLB and has raised the RCS iodine concentration to 60 times the maximum equilibrium RCS Technical Specification concentration of 0.2 μCi/gm of dose equivalent (DE) I-131 (i.e., 12 μCi/gm of DE I-131). For the accident initiated iodine spike the reactor trip associated with the SLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 0.2 μCi/gm of DE I-131. The accident initiated iodine spike is calculated assuming the letdown flowrate of 40 gpm and 100% efficiency of iodine in the demineralizer beds.

The iodine activity concentration of the secondary coolant at the time the SLB occurs is assumed to be equivalent to the Technical Specification limit of 0.1 μCi/gm of DE I-131.

The amount of primary to secondary SG tube leakage in the intact SG is assumed to be equal to the Technical Specification limit of 150 gpd. The allowable leak rate for the faulted SG was determined to be 9.0 gpm based on a density of 62.4 lbm/ft^3.

No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The SG connected to the broken steamline is assumed to boil dry within the initial 15 minutes following the SLB. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. Also, iodine carried over to the faulted SG by SG tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

An iodine partition factor in the intact SG of 0.01 (curies I /gm steam)/(curies I/gm water) is used (Reference 1).

Eight hours after the accident, the RHR System is assumed to be placed into service for heat removal, and there are no further steam releases to atmosphere from the secondary system.

The thyroid dose conversion factors, breathing rates, and atmospheric dispersion factors used in the dose calculations are given in Table 1. The parameters associated with the control room HVAC modes are summarized in Table 2. The remaining major assumptions and parameters used specifically in this analysis are itemized in Table 3.

Control Room Model

The Kewaunee control room HVAC system operates in one of two modes. Mode 1 is the normal HVAC mode, in which 2,500 cfm of air flow is outside air and 13,450 cfm is recirculated air all of which is unfiltered. Mode 2, which consists of 100% recirculated air within the control room a portion of which is filtered. In Mode 2 there is 200 cfm of unfiltered in-leakage. The parameters associated with the control room HVAC modes are summarized in Table 2.

For the steam line break accident it is assumed that the HVAC system begins in Mode 2 since the time required to shift to Mode 2 is short.

Description of Analyses Performed

The analysis of the steam line break (SLB) radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 1). Both the pre-accident iodine spike and accident initiated iodine spike models are analyzed for these release paths.

Acceptance Criteria

The offsite dose limits for a SLB with a pre-accident iodine spike are the guideline values of 10CFR100. These guideline values are 300 rem thyroid. For a SLB with an accident initiated iodine spike the acceptance criteria are a "small fraction of" the 10CFR100 guideline values, or 30 rem thyroid. The criteria defined in SRP Section 6.4 (Reference 2) will be used for the control room dose limit: 30 rem thyroid.

Results

The offsite and control room thyroid doses due to the SLB are given in Table 4.

Conclusions

It was determined that for a primary to secondary leak rate in the faulted SG of less than or equal to 9.0 gpm (based on a density of 62.4 ft^3) the offsite and control room thyroid doses are within the current NRC acceptance criteria for a steamline break accident.

References

1. NUREG-0800, Standard Review Plan 15.1.5, Appendix, A, "Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR, Rev. 2, July 1981.

2. NRC SRP Section 6.4, "Control Room Habitability System", Rev 2, July 1981, NUREG-0800.

TABLE I
DOSE CONVERSION FACTORS, BREATHING RATES AND ATMOSPHERIC DISPERSION FACTORS

Isotope	Thyroid Dose Conversion Factors (1) (rem/curie)
1-131	1.07 E6
1-132	6.29 E3
I-133	1.81 E5
I-134	1.07 E3
1-135	3.14 E4

Time Period (hr)	Breathing Rate (2) (m^3/sec)
0-8	3.47 E-4

	Atmospheric Dispersion Factors (sec/m^3)
Site Boundary	
0-2 hr	2.232E-4 (3)
Low Population Zone	
0-2 hr	3.977E-5 (3)
2-8 hr	4.100E-6 (3)
Control Room	
0-8 hr	2.930E-3 (4)

(1) ICRP Pub 30
(2) Regulatory Guide 1.4
(3) KNPP USAR Section 14 Table 14.3-8
(4) KNPP Updated Control Room Habitability Evaluation Report

TABLE 2

CONTROL ROOM PARAMETERS

Volume	127,600 ft^3
Total Flow Rate	
Mode 2	15,850 cfm
Unfiltered Makeup/In-leakage	
Mode 2	200 cfm
Filtered Makeup	
Mode 2	0 cfm
Filtered Recirculation	
Mode 2	2500 cfm
Filter Efficiency	
Elemental	90%
Organic	90%
Particulate	90%
Occupancy Factors	
0-8 hours	1.0

TABLE 3
ASSUMPTIONS USED FOR SLB DOSE ANALYSIS

Power	1721.4 MWt
Reactor Coolant Iodine Activity Prior to Accident	
Pre-Accident Spike	12 μCi/gm of DE I-131
Accident Initiated Spike	0.2 μCi/gm of DE I-131
Reactor Coolant Iodine Activity Increase Due to Accident Initiated Spike	500 times equilibrium release rate from fuel
Secondary Coolant Activity Prior to Accident	0.1 μCi/gm of DE I-131
SG Tube Leak Rate During Accident (Based on a Density of 62.4 ft^3)	
Intact SG	150 gpd
Faulted SG	9.0 gpm
Iodine Partition Factor	
Faulted SG	1.0 (SG assumed to steam dry)
Intact SG	0.01
Duration of Activity Release Secondary System	8 hours
Offsite Power	Lost
Steam Release from Intact SG	290,000 lb (0-2 hr) 433,254 lb (2-8 hr)

TABLE 4
SLB OFFSITE & CONTROL ROOM DOSES

	Dose	Acceptance Criteria
Site Boundary (0-2 hr)		
Thyroid: Accident Initiated Spike	2.62 rem	30 rem
Thyroid: Pre-Accident Spike	3.71 rem	300 rem
Low Populatiou Zone (0-8 hr)		
Thyroid: Accident Initiated Spike	1.05 rem	30 rem
Thyroid: Pre-Accident Spike	0.83 rem	300 rem
Control Room (0-8 hr)		
Thyroid: Accident Initiated Spike	29.47 rem	30 rem
Thyroid: Pre-Accident Spike	12.30 rem	30 rem