TXU

TXU Energy
Comanche Peak Steam
Electric Station
P.O. Box 1002 (E01)
Glen Rose, TX 76043
Tel: 254 897 8920
Fax: 254 897 6652
lance.terry@txu.com

C. Lance Terry
Senior Vice President &
Principal Nuclear Officer

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File # 00236

April 8, 2002

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

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION (CPSES)
DOCKET NOS. 50-445 AND 50-446
LICENSE AMENDMENT REQUEST (LAR) 02-05
REVISION TO TECHNICAL SPECIFICATION (TS) 3.4.16
REACTOR COOLANT SYSTEM LIMIT FOR DOSE
EQUIVALENT I-131

Gentlemen:

Pursuant to 10CFR50.90, TXU Generation Company LP (TXU Energy) hereby requests an amendment to the CPSES Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications. This change request applies equally to both units.

The proposed change will revise TS 3.4.16, "Reactor Coolant System Specific Activity." TS 3.4.16 limits the specific activity of Dose Equivalent I-131 (DEI) and the gross specific radioactivity of the reactor coolant. TXU Energy proposes to lower the Limiting Condition For Operation (LCO) for DEI in the Reactor Coolant System (RCS) from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm.

This submittal also requests approval of proposed changes to the main steam line break (MLSB) post-accident radiological dose consequences analysis that was previously approved for implementing the CPSES Steam Generator Alternate Repair Criteria (SG ARC).

The purpose of the LAR is as follows: (1) address an iodine appearance rate issue identified in CPSES post-accident thyroid dose calculations which are controlled by a postulated design basis accident initiated concurrent with an iodine spike; and (2) maintain the maximum benefit allowable after implementing the SG ARC on CPSES

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Unit 1. The iodine appearance rate issue is discussed in Westinghouse Nuclear Safety Advisory Letter NSAL-00-004 "Nonconservatisms in Iodine Spiking Calculations." This proposed TS change results from TXU Energy's evaluation of NSAL-00-004 and the benefits allowed after implementing the CPSES SG ARC.

Attachment 1 provides a detailed description of the proposed change, a safety analysis of the proposed change, TXU Energy's determination that the proposed change does not involve a significant hazard consideration, a regulatory analysis of the proposed change, and an environmental evaluation. Attachment 2 provides the affected Technical Specification pages marked-up to reflect the proposed change. Attachment 3 provides proposed changes to the Technical Specification Bases for information only. These changes will be processed per CPSES site procedures. Attachment 4 provides retyped Technical Specification pages which incorporate the requested changes. Attachment 5 provides retyped Technical Specification Bases pages which incorporate the proposed changes.

TXU Energy requests approval of the proposed License Amendment by March 31, 2003, to be implemented within 60 days. The approval date was administratively selected to allow for NRC review but CPSES does not require this amendment to allow continued safe full power operations. TXU Energy has already implemented interim administrative controls, consistent with this LAR, for CPSES Units 1 and 2 to limit the LCO for RCS DEI specific activity to 0.45 μCi/gm.

In accordance with 10CFR50.91(b), TXU Energy is providing the State of Texas with a copy of this proposed amendment.

This communication contains no new or revised commitments.

Should you have any questions, please contact Mr. Connie Wilkerson at (254) 897-0144

Enclosed is a copy of Westinghouse Letter NSAL-00-004 for your information.

I state under penalty of perjury that the foregoing is true and correct.

Executed on April 8, 2002.

Sincerely,

TXU Generation Company LP

By: TXU Generation Management Company LLC
Its General Partner

C. L. Terry
Senior Vice President and Principal Nuclear Officer

By: Roger D. Walker
Roger D. Walker
Regulatory Affairs Manager

CLW/clw

Attachments
1. Description and Assessment
2. Markup of Technical Specifications pages
3. Markup of Technical Specifications Bases pages (for information)
4. Retyped Technical Specification Pages
5. Retyped Technical Specification Bases Pages (for information)

Enclosures
1. Westinghouse Nuclear Safety Advisory Letter NSAL-00-004 "Nonconservatisms in Iodine Spiking Calculations," dated March 7, 2000

c - E. W. Merschoff, Region IV
W. D. Johnson, Region IV
D. H. Jaffe, NRR
Resident Inspectors, CPSES

Mr. Authur C. Tate
Bureau of Radiation Control
Texas Department of Public Health
1100 West 49th Street
Austin, Texas 78704

ATTACHMENT 1 to TXX-02047

DESCRIPTION AND ASSESSMENT

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1.0 DESCRIPTION

TXU Generation Company LP (TXU Energy) requests an amendment to the Comanche Peak Steam Electric Station (CPSES) Unit 1 Operating License (NPF-87) and CPSES Unit 2 Operating License (NPF-89) by incorporating the attached change into the CPSES Unit 1 and 2 Technical Specifications (TS). Proposed change LAR-02-05 is a request to revise TS 3.4.16, "Reactor Coolant System Specific Activity."

This submittal also requests approval of proposed changes to the main steam line break (MLSB) post-accident radiological dose consequences analysis that was previously approved for implementing the CPSES Steam Generator Alternate Repair Criteria (SG ARC).

Changes to the CPSES Final Safety Analysis Report (FSAR), consistent with this LAR, are being processed and are to be submitted to the NRC in FSAR Amendment 98 (August, 2002).

2.0 PROPOSED CHANGE

The proposed change revises TS 3.4.16 to lower the Limiting Condition For Operation (LCO) for Dose Equivalent Iodine-131 in the reactor coolant system from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm. The requested change is proposed after TXU Energy's evaluation of identified non-conservative inputs related to the plant accident dose analysis. Specifically, the requested change addresses two issues: (1) resolves a non-conservative determination of equilibrium iodine appearance rates identified in plant post-accident thyroid dose calculations which are controlled by a postulated design basis accident initiated concurrent with an iodine spike; and (2) maintains the existing maximum benefit allowable after implementing the Comanche Peak Steam Generator Alternate Repair Criteria on CPSES Unit 1.

The proposed changes to the MSLB accident analysis that supports the implementation of the CPSES SG ARC are due to an updated calculation of the radiological dose consequences that result from assuming a reduced TS 3.4.16 limit on equilibrium Dose Equivalent Iodine-131 of 0.45 μCi/gm and incorporating other conservative parameters as recommended by Westinghouse in NSAL-00-004 (See Section 4.0).

For Information only, this LAR includes associated changes to TS Bases B3.4.16. The current TS Bases B3.4.16 has been revised to provide a note that the current TS 3.4.16 limit for Dose Equivalent Iodine -131 of 1.0 μCi/gm is not conservative and that applicable plant analyses have been assessed at a limit of 0.45 μCi/gm and have been found acceptable. In accordance with NRC Administrative Letter 98-010, interim administrative controls have been enacted to prevent operation above 0.45 μCi/gm.

3.0 BACKGROUND

The CPSES Reactor Coolant System (RCS) is described in FSAR Section 5.0 and related design basis accidents (DBAs) are analyzed in FSAR Section 15. The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor. TS 3.4.16 limits the concentration of radionuclides allowed in the reactor coolant. Specifically, TS 3.4.16 limits the specific activity of Dose Equivalent Iodine-131 (DEI) and the gross specific radioactivity of the reactor coolant. These TS limits ensure that the radiological consequences of analyzed transients and accidents meet the requirements of 10 CFR 50 Appendix A, Criterion 19, for individuals in the Control Room, and are within applicable dose guideline limits per 10 CFR 100 for individuals that may be exposed offsite following a radioactive release.

This amendment request is proposed following TXU Energy's evaluation of Westinghouse Nuclear Safety Advisory Letter NSAL-00-004 "Nonconservatisms In Iodine Spiking Calculations" (Reference 1). NSAL-00-004 was issued to all Westinghouse plants after a Westinghouse licensee reported the issue to the NRC in a Licensee Event Report (Reference 2) and as a potential issue under 10 CFR Part 21. The NSAL involves the basic plant component--accident dose analysis, and it identifies an issue regarding the use of non-conservative assumptions in the calculation of accident-initiated iodine spike rates in the RCS. The correction of the non-conservative assumptions used in the historical analysis results in an increase in equilibrium iodine appearance rates in the reactor coolant that in turn results in an increased dose consequence for those DBAs that model the iodine spike rates.

NSAL-00-004 concluded that the issue did not represent a substantial safety hazard for Westinghouse plants given the overall conservative nature of the accident dose analysis model, e.g., plants do not routinely operate near the assumed accident dose analysis model inputs of maximum TS allowed reactor coolant leakage (typically 11 gpm -- 1 gpm unidentified + 10 gpm identified) and at the LCO for DEI in the RCS (typically 1.0 μCi/gm); however, the NSAL recommended that licensees perform a plant-specific review to ensure that affected DBA radiological consequence analysis assumptions are conservative and TS limits continue to be met. In particular, for plants which are using a Steam Generator Alternate Repair Criteria (SG ARC) to allow continued operation with identified steam generator tube indications, the NSAL recommended evaluation of the restrictions on the allowable indication level and/or the allowable primary coolant equilibrium iodine activity.

4.0 TECHNICAL ANALYSIS

TXU Energy's evaluation of the NSAL-00-004 issue determined that the basic iodine appearance rates used in the original analysis of CPSES DBA radiological consequences were obtained from Westinghouse in WPT-8945 (Reference 3) and were non-conservative in that they assumed non-conservative operating parameter inputs. As identified in NSAL-00-004, the parameter inputs that could be non-conservative and open to correction in an updated determination of equilibrium iodine appearance rates are:

-Letdown flow rate
-Letdown demineralizer iodine removal efficiency
-Primary coolant leakage
-Uncertainty in the letdown flow rate
-Primary coolant mass

TXU Energy's evaluation of the equilibrium iodine appearance rates determined that the original assumed inputs for RCS letdown flow rate (75 gpm), letdown demineralizer iodine removal efficiency (0.9), and primary coolant leakage (none assumed) are non-conservative. These operational inputs were corrected to reflect the more conservative assumptions recommended in NSAL-00-004 and then used to calculate updated equilibrium appearance rates for the iodine isotopes in the reactor coolant. A revised radiological dose consequence analysis was then performed for the affected CPSES DBAs using the updated equilibrium iodine appearance rates and associated concurrent iodine spike rates.

Impact on CPSES Design Basis Accident Analyses

TXU Energy has evaluated the NSAL-00-004 issue by reviewing the affected CPSES DBA analyses. This effort included developing updated equilibrium iodine appearance rates and iodine spike rates for the design-basis Steam Generator Tube Rupture (SGTR) event, Main Steam Line Break (MSLB) event, Reactor Coolant Pump Locked Rotor (RCPLR) event, Control Rod Ejection (CRE) event, and Small Line Break Outside Containment (SLBOC) event. The evaluation of these DBA analyses assumes an equilibrium RCS DEI limit of 0.45 μCi/gm (the proposed CPSES TS 3.4.16 limit).

For the SGTR, RCPLR, CRE and SLBOC events, TXU Energy's evaluation determined that correcting the non-conservative operating inputs and recalculating the iodine appearance/iodine spike rates for the current TS 3.4.16 limit on equilibrium RCS DEI activity (1.0 μCi/gm) would increase the radiological dose consequences to the thyroid but not significantly challenge the applicable DBA acceptance criteria, i.e., 30 rem thyroid to an individual in the control room or at the site boundary. Furthermore, TXU Energy's evaluation compared the existing DBA iodine spike rates which are based on the historical, non-conservative operating inputs and an equilibrium RCS DEI limit of 1.0 μCi/gm to updated iodine spike rates which are based on the conservative operating inputs recommended in NSAL-00-004 and a reduced equilibrium RCS DEI limit of 0.45 μCi/gm--the proposed TS 3.4.16 limit. The comparison showed that the existing DBA analyses for the SGTR, RCPLR, CRE and SLBOC events would be conservative with respect to the calculated radiological dose consequences to the thyroid; therefore, TXU Energy has not changed the dose analyses for these DBAs.

For the specific MSLB event supporting the CPSES SG ARC, TXU Energy's evaluation determined that correcting the non-conservative operating inputs and recalculating the iodine appearance/iodine spike rates for the current TS 3.4.16 limit on equilibrium RCS DEI activity (1.0

μCi/gm) would challenge the applicable DBA acceptance criteria without taking a compensating action to either increase restrictions on the allowable SG tube indication level and the number of indications or reduce the TS 3.4.16 limit on allowed equilibrium RCS DEI activity for continued operation. To retain the maximum benefits of the existing approved SG ARC primary-to-secondary leakage criteria, TXU Energy chose (as recommended in the NSAL-00-004) to reduce the equilibrium RCS DEI activity concentration limit from 1.0 μCi/gm to 0.45 μCi/gm and revise the associated DBA analysis for the MSLB event supporting the CPSES SG ARC, accordingly.

A further discussion and supporting data for these evaluations is given below.

Revised Iodine Appearance Rates

A station calculation shows the revised iodine appearance rates that incorporate the conservative recommended assumptions of NSAL-00-004 as well as the proposed TS 3.4.16 RCS DEI limit of 0.45 μCi/gm.

The basic equation used to compute the appearance rate of each important iodine isotope is:

$$P_i = 10^{-6} \frac{A_i C_1}{\Sigma A_i C_i} [\lambda_i M + \rho(EF + L)]$$

where

P_i = production rate of the *i*th isotope from defective fuel (atoms/sec)
A_i = the design-basis normal maximum concentration of the *i*th iodine isotope in the reactor coolant system (μci/gm)
C_1 = dose conversion factor for I-131 (mrem/μci)
C_i = dose conversion factor for the *i*th iodine isotope (mrem/μci)
λ_i = radioactive decay constant for the *i*th isotope (/sec)
M = reactor coolant mass (gm)
ρ = reactor coolant density (gm/cm^3)
E = purification efficiency (fraction)
F = letdown flow rate (cm^3/sec)
L = RCS leakage rate (cm^3/sec)

In the revised calculation, the following parameters are used to assure maximization of the equilibrium iodine appearance rates:

Nuclear Data:

Table I. Isotopic Dependent Input Data

Isotope	C_i[a] (mrem/µci)	A_i[b] (µci/gm)	λ_i[c] (/sec)
I-131	1.485E+3	2.8	9.9473E-7
I-132	5.353E+1	2.8	8.4300E-5
I-133	3.970E+2	4.2	9.2568E-6
I-134	2.537E+1	0.57	2.2089E-4
I-135	1.235E+2	2.3	2.8737E-5

a. C_i, Dose conversion factors from Table III of TID-14844
b. A_i, 1% failed fuel design basis activities
c. λ_i, Radioactive decay constants based on half lives from the Radiological Health Handbook

Operating Parameter Inputs:

-Letdown flow rate (F) = 140 gpm = 8833 cm^3/sec (conservative)
-Letdown demineralizer iodine removal efficiency (E) =1.0 (conservative)
-Primary coolant leakage (L) = 11 gpm = 694 cm^3/sec (conservative)
-Uncertainty in the letdown flow rate (assumed in value for letdown flow rate)
-Primary coolant mass (M) = 2.29E+08 gm
-Reactor coolant density (ρ) = 1.0 gm/cm^3 (conservative)

The updated equilibrium iodine appearance rates are listed in Table II.

Table II. Calculated Equilibrium Appearance Rates and Concurrent Iodine Spike Rates*

Isotope	Equilibrium Appearance Rate (Ci/sec)	Concurrent Iodine Spike Rate** (Ci/sec)
I-131	2.9093E-03	1.45
I-132	8.6110E-03	4.31
I-133	5.2122E-03	2.61
I-134	3.6561E-03	1.83
I-135	3.9495E-03	1.97

* Incorporates the conservative operating parameter inputs recommended in NSAL-00-004 and a RCS DEI activity limit of 0.45 µCi/gm.
** Concurrent Spike Rate = 500 x Equilibrium Appearance Rate

Impact on Design-Basis Accident Doses

To assess the impact on the affected CPSES DBA dose consequences, the updated concurrent iodine spike rates (shown in Table II) were compared to the original design basis concurrent iodine spike rates which are based on historical, non-conservative operating inputs and an assumed equilibrium RCS DEI limit of 1.0 μCi/gm --the current TS 3.4.16 limit. The adjustment ratio of the original to the updated concurrent iodine spike rate are provided in Table III for each iodine isotope.

Table III.

Ratio of Updated Concurrent Iodine Spike Rate to Design Basis Concurrent Iodine Spike Rate

Isotope	Design Basis Rate	Updated Rate	Ratio of Spike Rate
I-131	1.46	1.45	0.99
I-132	2.86	4.31	1.51
I-133	3.34	2.61	0.78
I-134	4.52	1.83	0.40
I-135	3.20	1.97	0.62

The thyroid dose assessment was performed by taking the sum of the updated rates times their isotopic dose conversion factor (listed in Table I) divided by the sum of the original rates times their isotopic dose conversion factor or,

$$\frac{\sum R_U C_i}{\sum R_D C_i}$$

Where:
R_U is the updated concurrent iodine spike rate of each isotope
R_D is the design basis concurrent iodine spike rate of each isotope
C_i is the dose conversion factor for each isotope

After applying the appropiate data from Tables I and III, the above equation results in a ratio of about 0.89 which is less than 1.0. This means that any thyroid dose calculated using the updated rate will be less than the existing design basis dose.

For gamma and beta doses, assessment of the impact of the updated iodine spike rates depends on the relative dose contribution from I-132 in comparison to that from other isotopes including noble gas daughter products. Owing to the fact that Xe-133m, Xe-133, Xe-135m, and Xe-135 are necessarily

reduced (since parents I-133 and I-135 have reduced appearance rates as shown in Table III), the following ratio may be assessed to assure that the I-132 contribution (which corresponds to an increased spike rate) is not controlling:

$$Ratio = \frac{1.51D_{I-132} + 0.78(D_{Xe-133m} + D_{Xe-133}) + 0.62(D_{Xe-135m} + D_{Xe-135})}{D_{I-132} + D_{Xe-133m} + D_{Xe-133} + D_{Xe-135m} + D_{Xe-135}}$$

where the "D" parameters represent the total accumulated isotopic dose contribution computed with the previous spike rates.

If the above ratio is less than 1.0, then it can be concluded that the total gamma or total beta dose obtained by application of the updated spike rates will be less than that obtained by use of the previous (design basis) spike rates.

The dose assessments described above have been applied to all potentially affected CPSES DBAs. For the SGTR, RCPLR, CRE and SLBOC events, it has been concluded that the proposed implementation of an equilibrium RCS DEI limit of 0.45 μCi/gm will ensure that the existing design basis dose consequences analyses that model the concurrent iodine spike methodology will be conservative; therefore, the existing accident dose consequences analyses for these DBAs are not being changed.

Impact on CPSES Steam Generator Alternate Repair Criteria (SG ARC)

Concurrent with the evaluation described above, TXU Energy evaluated the potential impacts on the CPSES SG ARC. TXU Energy implemented the CPSES SG ARC on Unit 1 effective September 30, 1999 in accordance with License Amendment No. 70 and the provisions of Generic Letter 95-05 "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes affected by Outside Stress Corrosion Cracking." The NRC approved TXU Energy's request for implementing the CPSES SG ARC in the Safety Evaluation issued with CPSES License Amendment No. 70 (Reference 4).

In approving the CPSES SG ARC the NRC accepted a specific DBA radiological consequences analysis for a bounding main steam line break (MSLB) event. The accepted radiological consequences analysis is described in the current CPSES FSAR, Chapter 15, and addresses two event permutations consistent with the Standard Review Plan (NUREG-0800): a pre-accident iodine spike (60 μCi/gm DEI in primary coolant) and an assumed accident-initiated iodine spike (500 times the equilibrium RCS activity of 1.0 μCi/gm DEI). The analysis was based on a maximum allowed accident-initiated leak rate of 27.79 gpm (as determined by the methodology prescribed in Section 2.b.4 of Generic Letter 95-05). Additionally, the analysis used dose conversion factors obtained from the International Commission on Radiation Protection Publication 30 (ICRP-30). The ICRP-30 dose conversion factors lead to more realistic thyroid dose results which are approximately thirty percent lower than would be obtained using factors from Table III of TID-14844"Calculation of Distance Factors for Power and Test Reactor Sites." Other key assumptions in the analysis are: The iodine activity concentration of the secondary coolant at the time the MSLB occurs is the TS limit of 0.1 μCi/gm DEI, and the amount of primary-to-secondary SG tube leakage in each intact SG is assumed to be equal to 150 gpd (the current TS limit).

A summary of the radiological dose consequences calculated for the CPSES SG ARC postulated MSLB event and previously accepted by the NRC is presented in Table IV.

Table IV. Calculated MSLB Doses Accepted by NRC for Implementing the CPSES SG ARC* (Current FSAR)

	Dose (rem)	Acceptance Criterion (rem)
EAB (0-2 hr)		
Thyroid: Pre-accident Spike	43.50	300
Thyroid: Accident initiated Spike	29.05	30
Whole Body	0.25	2.5
LPZ (0-8 hr)		
Thyroid: Pre-accident Spike	22.26	300
Thyroid: Accident initiated Spike	22.48	30
Whole Body	0.11	2.5
Control Room (0-8 hr)		
Thyroid: Pre-accident Spike	6.72	30
Thyroid: Accident initiated Spike	6.68	30
Whole Body	0.01	5

*Based on current TS 3.4.16 limit of 1.0 μCi/gm DEI

During evaluation of the NSAL-00-004 issue TXU Energy determined that the dose consequence analysis for the special bounding MSLB event used in approving the CPSES SG ARC was subject to the same non-conservatisms as discussed above for the other affected DBAs. TXU Energy's evaluation showed that correcting the non-conservative operating inputs and calculating updated iodine appearance rates for an equilibrium RCS DEI limit of 1.0 μCi/gm would challenge the applicable acceptance criteria for postulated DBA radiological dose consequences without a compensating action to increase restrictions on the allowable SG tube indication level and the number of indications. To avoid or mitigate this penalty, allow continued operation with known SG tube indications and retain the benefits of the existing SG ARC primary-to-secondary leakage criteria, the RCS DEI activity concentration must be limited to a value below the current TS limit of 1.0 μCi/gm.

Westinghouse performed a revised calculation in WPT-16126 (Reference 5) for the CPSES SG ARC which provided updated dose consequence analysis results. The revised analysis addressed the conservative operating parameter inputs as recommended by NSAL-00-004 which necessitated the

incorporation of a reduced equilibrium RCS DEI limit of 0.45 μCi/gm to maintain an equivalent SG ARC benefit. The updated iodine spike rates used in the revised Westinghouse MSLB dose analysis for the CPSES SG ARC are as follows:

I-131 103.2 Ci/min
I-132 304.3 Ci/min
I-133 185.0 Ci/min
I-134 128.6 Ci/min
I-135 140.7 Ci/min

In the revised calculation provided by Westinghouse, the control room dose calculations were extended to account for the contribution due to continued breathing of radioactivity in the control room after the releases are terminated. Although the control room doses increase as a result, this does not impose any limitations on plant operation as there is significant margin to the limits. All other aspects of the SG ARC dose consequence analysis methodology remained the same. The results of the revised dose consequence analysis for the CPSES SG ARC is provided in Table V.

Table V. Updated MSLB Doses for Implementing the CPSES SG ARC*

	Dose (rem)	Acceptance Criterion (rem)
EAB (0-2 hr)		
Thyroid: Pre-accident Spike	43.50	300
Thyroid: Accident initiated Spike	26.88	30
Whole Body	0.15	2.5
LPZ (0-8 hr)		
Thyroid: Pre-accident Spike	22.26	300
Thyroid: Accident initiated Spike	21.45	30
Whole Body	0.10	2.5
Control Room (0-8 hr)		
Thyroid: Pre-accident Spike	7.45	30
Thyroid: Accident initiated Spike	7.16	30
Whole Body	0.01	5
Beta Skin	1.5	30

*Based on using updated iodine appearance rates assuming an equilibrium RCS DEI limit of 0.45 μCi/gm and incorporating the conservative operating inputs recommended by NSAL-00-004

The updated dose consequence results for the CPSES SG ARC in Table V are approximately equivalent to the dose consequences of the existing analysis accepted by the NRC (Table IV). Furthermore, the off-site doses are reduced by the proposed implementation of a 0.45 μCi/gm RCS DEI limit. Although the Control Room doses increase slightly, they are still below the applicable DBA acceptance criteria for control room habitability and therefore do not impact the design or licensing basis. The proposed implementation of a RCS DEI limit of 0.45 μCi/gm will ensure that: (1) CPSES maintains the full benefit of the previously established SG ARC benefits; and (2) CPSES MSLB dose consequences will remain below the regulatory guideline values.

Summary of Impacts on Affected CPSES Design Basis Accidents

TXU Energy's evaluation of the the NSAL-00-004 issue determined that the existing CPSES DBA analyses for the SGTR, RCPLR, CRE and SLBOC events were affected by the non-conservative input assumptions as identified in the NSAL. The existing radiological dose consequences associated with these DBA's were underestimated. TXU Energy corrected the non-conservative assumptions in the analysis inputs and developed updated equilibrium iodine appearance rates and iodine spike rates based on an RCS DEI activity limit of 0.45 μCi/gm, and then evaluated the corresponding radiological dose consequences for the events. The evaluation showed that the radiological dose consequences resulting from the updated DBA analyses to individuals in the CR and at the EAB/EPZ would be less than the dose consequences of the existing design basis analyses, and remain less than the applicable regulatory guidance limits. Therefore, the existing radiological dose consequences analyses for the SGTR, RCPLR, CRE and SLBOC are not being changed.

TXU Energy's evaluation of the existing CPSES SG ARC determined that the supporting MSLB analysis was also affected by the non-conservative input assumptions as identified in the NSAL. The existing radiological dose consequences associated with this MSLB event were underestimated if the maximum benefit of the SG ARC is to be maintained. TXU Energy corrected the non-conservative assumptions in the analysis inputs and, to continue to maintain the maximum benefit of the current SG ARC analysis, elected to reduce the analysis limit for equilibrium RCS DEI activity from 1.0 μCi/gm to 0.45 μCi/gm. With these changes in the analysis, updated equilibrium iodine appearance rates and iodine spike rates were developed and used to calculate updated radiological dose consequences for the MLSB event. The updated radiological dose consequences to individuals in the CR and at the EAB/EPZ (see Table V) are approximately equivalent to the existing licensing-basis values, (see Table IV) and remain less than the applicable regulatory guidance limits.

In summary, the proposed change to revise TS 3.4.16 to lower the Limiting Condition For Operation (LCO) for DEI in the reactor coolant system from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm allows continued safe operation because it is consistent with maintaining the existing benefits of the approved CPSES SG ARC analysis. Additionally, the proposed change to the DBA radiological dose analysis for the MSLB event supporting the CPSES SG ARC and all updated calculated doses assessed for the other affected DBAs at an assumed equilibrium RCS DEI activity limit of 0.45 μCi/gm remain a small fraction (10 percent) of the 10 CFR Part 100 standards. The control room doses meet the acceptance criteria in 10 CFR Part 50, Appendix A, General Design Criteria 19.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

TXU Generation Company LP has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10CFR50.92, "Issuance of amendment," as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change to revise Technical Specification (TS) 3.4.16 "Reactor Coolant System Specific Activity" to reduce the Limiting Condition For Operation (LCO) for Dose Equivalent I-131 in the reactor coolant from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm and the revised main steam line break (MSLB) radiological consequence analysis are used to determine post-accident dose. They are not related to any accident initiator. Therefore, this change cannot increase the probability of an accident.

   The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR Part 100 and 10 CFR Part 50, Appendix A Criterion 19 limits and the NUREG-0800 SRP section 15.1.5 and section 6.4 guideline values.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change to revise TS 3.4.16 "Reactor Coolant System Specific Activity" to reduce the LCO for Dose Equivalent I-131 in the reactor coolant from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm and the revised MSLB radiological consequence analysis do not involve any physical plant changes. The change does not involve changes in operation of the plant that could introduce a new failure mode for creating an accident or affect the mitigation of an accident.

   Therefore, the proposed change does not create the possibility of a new or different accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

The proposed change to revise TS 3.4.16 "Reactor Coolant System Specific Activity" to reduce the LCO for Dose Equivalent I-131 in the reactor coolant from a specific activity of 1.0 μCi/gm to 0.45 μCi/gm is a conservative change in that this reduced TS limit, when used in applicable plant radiological dose consequence analysis models with all other input parameters held constant, calculates decreased dose consequences to the thyroid. The change, with all other analysis input parameters held constant, increases the margin to acceptance limits. Therefore, this change does not result in a significant reduction in the margin provided by TS 3.4.16.

The revised MSLB offsite and control room radiological consequences analysis dose results are within 10 CFR Part 100 and 10 CFR Part 50, Appendix A Criterion 19 limits and the NUREG-0800 SRP section 15.1.5 and section 6.4 guideline values

Therefore the proposed change does not involve a reduction in a margin of safety.

Based on the above evaluations, TXU Generation Company LP concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10CFR50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The applicable regulatory requirements and guidance criteria related to this proposed LAR are as follows:

10CFR100.11 specifies the maximum dose to the whole body (25 rem) and thyroid (300 rem) that an individual at the site boundary can receive for 2 hours during an accident.

10CFR50 Appendix A, Criterion 19 specifies the maximum dose to the whole body (5 rem), or its equivalent to any part of the body, that an individual in the Control Room can receive for the duration of an accident.

CPSES TS 3.4.16 limits the concentration of radionuclides allowed in the reactor coolant. Specifically, TS 3.4.16 limits the specific activity of Dose Equivalent Iodine-131 (DEI) and the gross specific radioactivity of the reactor coolant. These TS limits ensure that the radiological consequences of analyzed transients and accidents meet the requirements of 10 CFR 50 Appendix A, Criterion 19, for individuals in the Control Room, and are within applicable dose guideline limits per 10 CFR 100 for individuals that may be exposed offsite following a radioactive release.

NUREG-0800 Standard Review Plan, Section 6.4 provides guidance on acceptance criteria with respect to control room habitability following an accident. Dose guideline limits refer to GDC 19 and are as follows for individuals : whole body (5 rem); thyroid (30 rem); beta skin dose (30 rem).

NUREG-0800 Standard Review Plan, Section 15.1.5 and 15.6.3 provide guidance on acceptance criteria with respect to the radiological consequences of certain analyzed design basis accidents. Acceptance criteria for offsite exposure limits are based on the requirements of 10CFR100. For accidents with an assumed pre-accident iodine spike in the RCS corresponding to the maximum allowable iodine concentration in the TS (typically 60 μCi/gm), calculated accident doses to an individual at the site boundary should not exceed the values of 10CFR100.11, i.e., 25 rem (whole body) and 300 rem (thyroid). For accidents with an assumed accident initiated iodine spike in the RCS with the reactor coolant at the maximum allowed equilibrium value for continued full power operation (typically 1.0 μCi/gm), calculated accident doses to an individual at the site boundary should not exceed "a small fraction" of the values in 10CFR100.11, i.e., 2.5 rem (whole body) and 30 rem (thyroid).

The above regulations apply to, and the guidance is consistent with, the CPSES siting and design requirements with respect to providing assurance that the radiological dose consequences from postulated accidents will be acceptably low for individuals remaining in the control room and individuals who may be exposed offsite at the site boundary. The specific dose limits given in these regulations/guidance documents serve as the acceptance criteria for the plant accident analyses.

TXU Energy's evaluation of the NSAL-00-004 issue has resulted in a re-analysis of affected DBAs as described above in Section 4.0. After incorporating the conservative assumptions recommended by the NSAL and assuming a reduced limit for equilibrium RCS DEI specific activity at 0.45 μCi/gm, all accident analyses were determined to be less than the regulatory guidance limits.

TXU Energy has implemented administrative controls for CPSES Units 1 and 2, consistent with this amendment request and NRC Administrative Letter 98-010, to further limit the TS 3.4.16 Limiting Condition For Operation (LCO) for RCS DEI specific activity to 0.45 μCi/gm. This action prevents operation above 0.45 μCi/gm, maintains the maximum benefit gained from implementing the current CPSES SG ARC on CPSES Unit 1, and ensures that related DBA radiological dose consequence analyses are acceptable. The current TS Bases B3.4.16 has been revised to provide a note that the current TS 3.4.16 limit of 1.0 μCi/gm is not conservative and that applicable plant analyses have been assessed at a limit of 0.45 μCi/gm and have been found acceptable.

In conclusion, based on the considerations discussed above, (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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

TXU Generation Company LP has determined that the proposed amendment would change the Technical Specifications limit for the maximum allowed equilibrium value of Dose Equivalent I-131 in the reactor coolant for continued full power operation and the radiological dose consequences analysis for the Main Steam Line Break event supporting the implementation of the Comanche Peak Steam Generator Alternate Repair Criteria. TXU Generation Company LP has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. NSAL-00-004. Westinghouse Nuclear Safety Advisory Letter "Nonconservatisms In Iodine Spiking Calculations" issued March 7, 2000 to all Westinghouse plants. (Enclosure 1)

2. Beaver Valley Power Station Unit 1 Licensee Event Report (LER) # 99-002-00 "Non-Conservative Concurrent Iodine Spike Radiological Dose Calculation Methodology" dated March 3, 1999

3. WPT-8945. Westinghouse Letter to TU Electric, CPSES Units 1 and 2, dated May 29, 1987

4 NRC Safety Evaluation by Office Of Nuclear Reactor Regulation Related to Amendment No. 70 to Facility Operating Licenses No. NPF-87 and NPF-89, dated September 22, 1999

5. WPT-16126. Westinghouse Letter to TXU Electric Company, CPSES Units 1 & 2 "Steamline Break Dose Analysis," dated August 3, 2000

ATTACHMENT 2 to TXX-02047

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Pages [3.4-44, 3.4-46]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > ~~1.0~~ 0.45 µCi/gm.	----------Note---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ ~~1.0~~ 0.45 μCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine E from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

ATTACHMENT 3 to TXX-02047

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)
(For Information Only)

Pages [B3.4-101 thru B3.4-106]

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref.1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or a main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are ~~standardized, based on parametric evaluations of offsite radioactivity dose consequences for typical site locations.~~ specific to CPSES due to the implementation of the alternate steam generator tube repair criteria.

~~The parametric evaluations showed the potential offsite dose levels for a SGTR accident were an appropriately small fraction of the 10 CFR 100 dose guideline limits. Each evaluation assumes a broad range of site applicable atmospheric dispersion factors in a parametric evaluation.~~

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR or a MSLB accident. The SGTR safety analysis (Ref.2) assumes the specific activity of the reactor coolant at the LCO limit[1] and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The MSLB safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 27.8 gpm[2] in the affected steam generator and 450 gpd combined in the unaffected steam generators. The safety analysis for both accidents assumes the specific activity of the secondary coolant at its limit of 0.1 µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

(continued)

[1]The referenced safety analysis reports the doses for the SGTR assuming DEI-131 to be 1.0 µCi/gm. The doses reported using this value have been shown to be conservative relative to those that would be calculated using a DEI-131 value of 0.45 µCi/gm.

[2]To obtain the maximum benefit from the steam generator alternate repair criteria, the MSLB radiological consequences analysis assumes a leak rate to the faulted steam generator during the accident that results in calculated consequences approaching a small fraction (10%) of the 10 CFR 100 guideline values for the accident initiated spike. This leak rate provides a maximum primary-to-secondary leak rate limit against which the predicted end-of-cycle leakage is compared.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the ~~SGTR~~ MSLB accident establishes the acceptance limits for RCS specific activity. However, the SGTR accident analysis consequences are significant. Reference to these analyses are ~~this analysis is~~ used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

~~The analysis is for~~ Each of the above analyses must consider two cases of reactor coolant specific activity. One case assumes specific activity at ~~1.0~~ 0.45 µCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 500 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 µCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/$\bar{E}$ µCi/gm for gross specific activity.

~~The analysis~~ These analyses also assume~~s~~ a loss of offsite power at the same time as the SGTR or the MSLB event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature N-16 signal. The MSLB causes a reduction in reactor coolant temperature and pressure. The temperature decrease causes an increase in reactor power. The power increase will trip the reactor on high neutron flux or overpower N-16. The pressure decrease will initiate a reactor trip on either low pressurizer pressure or the safety injection signal initiated by low pressurizer pressure, low steam generator pressure, or high containment pressure.

For the SGTR and the MSLB,~~T~~ the coincident loss of offsite power causes the steam dump valves to close to protect the condenser. For the SGTR, a ~~The~~ rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric relief valves and the main steam safety valves. A failure to close of the atmospheric relief valve on the affected SG is also assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends. For the MSLB, an uncontrolled (i.e, released to atmosphere) blowdown of only one steam generator is assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends. Radioactively contaminated steam is released to the atmosphere through the faulted SG as well as the intact SGs assuming the primary to secondary leak rates shown above.

The applicable safety analysis shows the radiological consequences of ~~an~~ either an SGTR or an MSLB accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 µCi/gm DOSE EQUIVALENT I-131.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR or an MSLB accident occurring during the established 48 hour time limit. The occurrence of an SGTR or an MSLB accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to ~~1.0~~ 0.45 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be within the allowed thyroid dose guideline. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref.2) and the MSLB accident analysis (Ref. 3) ~~shows~~ show that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR and an MSLB to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the offsite release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves. For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the offsite release of radioactivity in the event of an MSLB is unlikely. This is because the saturation pressure of the reactor coolant is sufficiently low that the primary-to-secondary pressure difference created when the affected SG depressurizes is not large enough to cause significant SG tube damage. Also, for the unaffected SGs the offsite release of radioactivity in the event of an MSLB is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves.

ACTIONS

A.1 and A.2

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

~~NOTE~~

~~The LCO limit in the TS is 1.0 µCi/gm. However, it has been identified that this limit is not conservative. All applicable analyses have been assessed at a limit of 0.45 µCi/gm and have been found to be acceptable. In accordance with NRC Administrative Letter 98-010, interim administrative controls have been enacted to prevent operation above 0.45 µCi/gm. Therefore an administrative limit of 0.45 µCi/gm shall apply while a license amendment is being processed (LDCR TS-2001-005).~~ 16

(continued)

BASES

ACTIONS (continued)

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply .

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant to an acceptable level. ~~below the setpoints of the main steam safety valves and prevents venting the SG to the environment in an SGTR event.~~ The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 10 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} greater than or equal to 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 10 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. FSAR, Section 15.6.3.

3. FSAR, Section 15.1.5

ATTACHMENT 4 to TXX-02047

RETYPED TECHNICAL SPECIFICATION PAGES

Pages [3.4-44, 3.4-46]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

LCO 3.4.16 The specific activity of the reactor coolant shall be within limits.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) ≥ 500°F

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 > 0.45 µCi/gm.	----------Note---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Verify DOSE EQUIVALENT I-131 within the acceptable region of Figure 3.4.16-1.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	48 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.16.2	----------NOTE---------- Only required to be performed in MODE 1. ---------- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 0.45 µCi/gm.	14 days AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	----------NOTE---------- Not required to be performed until 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. ---------- Determine E from a sample taken in MODE 1 after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours.	184 days

ATTACHMENT 5 to TXX-02047

RETYPED TECHNICAL SPECIFICATION BASES PAGES
(For Information Only)

Pages [B3.4-101 thru B3.4-106]

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.16 RCS Specific Activity

BASES

BACKGROUND

The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref.1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) or a main steam line break (MSLB) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and gross specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits. The limits in the LCO are specific to CPSES due to the implementation of the alternate steam generator tube repair criteria.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensures that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following a SGTR or a MSLB accident. The SGTR safety analysis (Ref.2) assumes the specific activity of the reactor coolant at the LCO limit[1] and an existing reactor coolant steam generator (SG) tube leakage rate of 1 gpm. The MSLB safety analysis (Ref. 3) assumes the specific activity of the reactor coolant at the LCO limit and an existing reactor coolant steam generator (SG) tube leakage rate of 27.8 gpm[2] in the affected steam generator and 450 gpd combined in the unaffected steam generators. The safety analysis for both accidents assumes the specific activity of the secondary coolant at its limit of 0.1 µCi/gm DOSE EQUIVALENT I-131 from LCO 3.7.18, "Secondary Specific Activity."

(continued)

[1]The referenced safety analysis reports the doses for the SGTR assuming DEI-131 to be 1.0 µCi/gm. The doses reported using this value have been shown to be conservative relative to those that would be calculated using a DEI-131 value of 0.45 µCi/gm.

[2]To obtain the maximum benefit from the steam generator alternate repair criteria, the MSLB radiological consequences analysis assumes a leak rate to the faulted steam generator during the accident that results in calculated consequences approaching a small fraction (10%) of the 10 CFR 100 guideline values for the accident initiated spike. This leak rate provides a maximum primary-to-secondary leak rate limit against which the predicted end-of-cycle leakage is compared.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The analysis for the MSLB accident establishes the acceptance limits for RCS specific activity. However, the SGTR accident analysis consequences are significant. Reference to these analyses are used to assess changes to the unit that could affect RCS specific activity, as they relate to the acceptance limits.

Each of the above analyses must consider two cases of reactor coolant specific activity. One case assumes specific activity at 0.45 µCi/gm DOSE EQUIVALENT I-131 with a concurrent large iodine spike that increases the I-131 activity in the reactor coolant by a factor of about 500 immediately after the accident. The second case assumes the initial reactor coolant iodine activity at 60.0 µCi/gm DOSE EQUIVALENT I-131 due to a pre-accident iodine spike caused by an RCS transient. In both cases, the noble gas activity in the reactor coolant assumes 1% failed fuel, which closely equals the LCO limit of 100/Ē µCi/gm for gross specific activity.

These analyses also assume a loss of offsite power at the same time as the SGTR or the MSLB event. The SGTR causes a reduction in reactor coolant inventory. The reduction initiates a reactor trip from a low pressurizer pressure signal or an RCS overtemperature N-16 signal. The MSLB causes a reduction in reactor coolant temperature and pressure. The temperature decrease causes an increase in reactor power. The power increase will trip the reactor on high neutron flux or overpower N-16. The pressure decrease will initiate a reactor trip on either low pressurizer pressure or the safety injection signal initiated by low pressurizer pressure, low steam generator pressure, or high containment pressure.

For the SGTR and the MSLB, the coincident loss of offsite power causes the steam dump valves to close to protect the condenser. For the SGTR, a rise in pressure in the ruptured SG discharges radioactively contaminated steam to the atmosphere through the SG atmospheric relief valves and the main steam safety valves. A failure to close of the atmospheric relief valve on the affected SG is also assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends. For the MSLB, an uncontrolled (i.e, released to atmosphere) blowdown of only one steam generator is assumed. The unaffected SGs remove core decay heat by venting steam to the atmosphere until the cooldown ends. Radioactively contaminated steam is released to the atmosphere through the faulted SG as well as the intact SGs assuming the primary to secondary leak rates shown above.

The applicable safety analysis shows the radiological consequences of either an SGTR or an MSLB accident are within a small fraction of the Reference 1 dose guideline limits. Operation with iodine specific activity levels greater than the LCO limit is permissible, if the activity levels do not exceed the limits shown in Figure 3.4.16-1, in the applicable specification, for more than 48 hours. The safety analysis has concurrent and pre-accident iodine spiking levels up to 60.0 µCi/gm DOSE EQUIVALENT I-131.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The remainder of the above limit permissible iodine levels shown in Figure 3.4.16-1 are acceptable because of the low probability of a SGTR or an MSLB accident occurring during the established 48 hour time limit. The occurrence of an SGTR or an MSLB accident at these permissible levels could increase the site boundary dose levels, but still be within 10 CFR 100 dose guideline limits.

The limits on RCS specific activity are also used for establishing standardization in radiation shielding and plant personnel radiation protection practices.

RCS specific activity satisfies Criterion 2 of 10CFR50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to 0.45 μCi/gm DOSE EQUIVALENT I-131, and the gross specific activity in the reactor coolant is limited to the number of μCi/gm equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the Design Basis Accident (DBA) will be within the allowed thyroid dose guideline. The limit on gross specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the DBA will be a small fraction of the allowed whole body dose.

The SGTR accident analysis (Ref.2) and the MSLB accident analysis (Ref. 3) show that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature ≥ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and gross specific activity are necessary to contain the potential consequences of an SGTR and an MSLB to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the offsite release of radioactivity in the event of a SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves. For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the offsite release of radioactivity in the event of an MSLB is unlikely. This is because the saturation pressure of the reactor coolant is sufficiently low that the primary-to-secondary pressure difference created when the affected SG depressurizes is not large enough to cause significant SG tube damage. Also, for the unaffected SGs the offsite release of radioactivity in the event of an MSLB is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety and relief valves.

ACTIONS

A.1 and A.2

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

With the DOSE EQUIVALENT I-131 greater than the LCO limit, samples at intervals of 4 hours must be taken to demonstrate that the limits of Figure 3.4.16-1 are not exceeded. The Completion Time of 4 hours is required to obtain and analyze a sample. Sampling is done to continue to provide a trend.

The DOSE EQUIVALENT I-131 must be restored to within limits within 48 hours. The Completion Time of 48 hours is required, if the limit violation resulted from normal iodine spiking.

(continued)

BASES

ACTIONS (continued)

B.1

With the gross specific activity in excess of the allowed limit, the unit must be placed in a MODE in which the requirement does not apply .

The change within 6 hours to MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant to an acceptable level. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 10 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} greater than or equal to 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 10 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. FSAR, Section 15.6.3.

3. FSAR, Section 15.1.5

ENCLOSURE 1 to TXX-02047

Westinghouse Nuclear Safety Advisory Letter NSAL-00-004
"Nonconservatisms in Iodine Spiking Calculations"

Nuclear Safety Advisory Letter

Westinghouse Electric Company

This is a notification of a recently identified potential safety issue pertaining to basic components supplied by Westinghouse. This information is being provided to you so that a review of this issue can be conducted by you to determine if any action is required.

P.O. Box 355, Pittsburgh, PA 15230

Subject: Nonconservatisms in Iodine Spiking Calculations	**Number:** NSAL-00-004
Basic Component: Accident Dose Analysis	**Date:** March 7, 2000
Plants: All Westinghouse Plants	

Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a)	Yes ☐	No ☐
Transfer of Information Pursuant to 10 CFR 21.21(b)	Yes ☐	
Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☒	

References: BVPS Unit 1 LER #: 99-002-00, 03/03/99, Non-Conservative Concurrent Iodine Spike Radiological Dose Calculation Methodology, (NRC Part 21 Log 99-15-1), http://www.nrc.gov/NRC/PUBLIC/PART21/1999/1999151.html.

SUMMARY

Nonconservative assumptions may have been used in the calculation of the accident-initiated iodine spiking rates in the primary coolant. A Westinghouse designed NSSS plant reported the issue to the NRC in a License Event Report (LER) and as a potential issue under 10 CFR Part 21.

The accident-initiated iodine spiking rates are derived from the equilibrium iodine appearance rates. Westinghouse has reviewed assumptions used in calculation of iodine appearance rates. This review has identified areas of nonconservatism in the historical approach for calculating iodine appearance rates. Since the accident-initiated iodine spiking rates are used in the determination of the radiological consequences of certain design basis accidents, correcting the nonconservatisms in the analysis may result in increases in the accident doses unless mitigating actions are taken. For plants which are using an Alternate Repair Criteria (ARC) to allow continued operation with identified steam generator tube indications, evaluation of the restrictions on the allowable indication level and/or the allowable primary coolant equilibrium iodine activity may be required.

Prior to any reanalysis of affected accident dose calculations, it is recommended that an administrative control be imposed to make certain that accident analysis assumptions and Technical Specification limits are met.

Additional information, if required, may be obtained from the originator. Telephone 412-374-4856.

Originator(s):

D. A. Lindgren
Regulatory & Licensing Engineering

H. A. Sepp, Manager
Regulatory & Licensing Engineering

TECHNICAL DESCRIPTION

ISSUE DESCRIPTION

Nonconservative assumptions may have been used by Westinghouse in the calculation of the accident-initiated iodine spiking rates in the primary coolant. The accident-initiated iodine spiking rates are derived from the equilibrium iodine appearance rates. The equilibrium iodine appearance rates are those at which the various iodine isotopes enter the primary coolant system from fuel elements having clading leaks and are balanced by cleanup, primary coolant losses by leakage, and radioactive decay so that the primary coolant activity is maintained in a rough equilibrium. Revising the nonconservative analysis assumptions will result in an increase in the equilibrium iodine appearance rates that will, in turn, result in increased doses for accidents that model the accident-initiated iodine spiking rates.

TECHNICAL EVALUATION

The assumptions used in calculation of equilibrium iodine appearance rates, especially those identified as being nonconservative, were reviewed. This review has revealed that concerns identified in the reference Licensee Event Report (LER) are valid. (Note, one of the concerns in the LER reflects a change in plant operation and not an initial nonconservatism.) The Westinghouse review also identified an additional area of nonconservatism in the historical approach for calculating equilibrium iodine appearance rates. Since the accident-initiated iodine spiking rates are used in the determination of the radiological consequences of certain design basis accidents (most notably steam generator tube rupture and main steam line break outside containment), correcting the nonconservatisms in the analysis may result in increases in the accident doses unless mitigating action is taken.

For plants which are using a steam generator Alternate Repair Criteria (ARC), there may be increased restrictions on the allowable indication level, number of indications, or the allowable primary coolant equilibrium iodine activity. Use of Alternate Repair Criteria allows continued operation with identified steam generator tube indications. Alternate Repair Criteria that may require evaluation include those that use a voltage based plugging criteria to address tube support plate elevation outside diameter stress corrosion cracking (ODSCC).

The following is a discussion of the areas identified as nonconservatisms in the equilibrium iodine appearance rate calculations.

- Letdown flow rate:

  When the original determination of equilibrium iodine appearance rates was made, there was an unstated assumption that the letdown would normally be operating with only one letdown orifice in service (this was the defined system normal operation). The maximum purification flow was understood as being intended for the period of time when the plant was approaching a planned shutdown; and it was desired to reduce the coolant activity level to as low as possible. (Note: It is also possible to open all three letdown orifices but this mode of operation is intended only to occur after shutdown while system pressure is being reduced; and it is desired to offset the reduction in pressure so as to maintain cleanup at as high a rate as possible.) Under this limited usage of elevated letdown flow, the high flow condition can be considered not to be applicable to the postulated accident - especially in the situation where there is both elevated letdown flow and elevated coolant activity.

  However, if a plant chooses to operate with higher letdown flow for normal operation (as opposed to limiting it to preparation for shutdown), the iodine appearance rates would be recalculated to reflect

the increase in the rate of cleanup. This higher cleanup rate permits the plant to operate with a higher fuel activity release level and still remain within its Technical Specification limit for primary coolant activity. Then, when an iodine spike occurs, the amount of iodine entering the primary coolant and the associated appearance rate would be greater because of the higher level of fuel activity release.

The above discussion is applicable to any plant choosing to operate at letdown flow rates in excess of that obtained with the one letdown orifice in operation.

- Letdown demineralizer iodine removal efficiency

  An iodine decontamination factor (DF) of 10 also was assumed in the determination of the equilibrium iodine appearance rates. However, it is more conservative to assume that the demineralizer has an infinite DF as it would be appropriate to assume that the demineralizer removes all iodine from the letdown flow. Thus, while use of a DF of 10 is a conservative assumption for calculating maximum coolant concentration, it is not conservative for determining the iodine appearance rates that are to be used in accident analysis.

  This discussion is applicable to all plants.

- Primary coolant leakage

  The original determination of the equilibrium iodine appearance rates did not take into consideration the impact of primary coolant leakage. There are three leakages recognized by the Technical Specifications that should be included in the calculation of iodine appearance rates in order to eliminate the nonconservatism from the calculation:

  1. Unidentified leakage (per typical Technical Specifications, this is $\leq$1.0 gpm)
  2. Identified leakage (per typical Technical Specifications, this is $\leq$10.0 gpm)
  3. Primary to secondary leakage (per typical Technical Specifications, this is 1.0 gpm total)

  Since the primary to secondary leakage is included in the identified leakage limit, the total leakage (identified plus unidentified) that should be considered is thus 11 gpm.

  This discussion is applicable to all plants, although the specific leakage limits for each plant should be confirmed.

- Uncertainty in the letdown flow rate

  Operating experience at plants makes it clear that the specified flow through the letdown flow path is a nominal rate and that the flow may be greater than the rated flow. Applying an uncertainty to the letdown (for example, up to ten percent) would address this nonconservatism.

  This discussion is applicable to all plants, although the size of the letdown flow rate uncertainty may be plant specific.

- Primary coolant mass

  The determination of equilibrium iodine appearance rates used a primary coolant mass that was based on a full pressurizer. Accordingly, the calculated primary coolant mass is slightly greater than the primary coolant mass actually present during operation. This represents a small conservatism. When the primary coolant mass is reduced to that of actual operation, the iodine appearance rates drop slightly.

This discussion is applicable to all plants. RCS volume needs to be a consistently applied value to bound the maximum dose.

CONCLUSIONS

Implementing the above changes in analysis assumptions will result in an increase in the equilibrium iodine appearance rates that will, in turn, result in increased doses for accidents that model the accident-initiated iodine spiking rates. Note, however, that if a plant is committed to using the maximum purification letdown flow only for short periods of time such as during the approach to a shutdown, it should not be necessary to base the iodine appearance rates on the higher flow rate.

Addressing the identified nonconservatisms in the existing determination of iodine appearance rates prior to reanalysis of doses requires reductions in the primary coolant iodine activity limit. These reductions are needed to assure that the accident analyses remain representative of potential operation.

There also may be modifications to the accident dose analysis assumptions that would reduce the impact of an increase in the appearance rate. These modifications include calculating thyroid doses using updated dose conversion factors and a less conservative iodine partition coefficient in the intact steam generators.

For plants using alternate repair criteria, implementing the identified changes in analytical assumptions may result in a reduction in the allowable value for the maximum allowable primary to secondary leak rate. This penalty can be avoided by modifying the Technical Specifications to specify a lower limit on the primary coolant iodine concentration, by administratively limiting the coolant concentration, by correcting calculated Dose Equivalent I-131 for actual letdown flow rate, or by reducing letdown flow rate to that assumed in the analysis. Some plants have already reduced the Technical Specification limit on primary coolant iodine concentration below the standard value of 1.0 μCi/g Dose Equivalent I-131 as part of the alternate repair criteria program. (Note that if a plant is already operating with a reduced Technical Specification limit for primary coolant iodine activity, there may not be margin for an additional reduction.)

ASSESSMENT OF SAFETY SIGNIFICANCE

As a result of the use of nonconservative assumptions in the calculation of the equilibrium iodine appearance rates and, in turn, the accident-initiated iodine spiking rates in the primary coolant, a question exists whether accident doses to the public and to the control room operators would remain within identified dose acceptance criteria.

The level of RCS iodine activity has a very strong influence in the calculated accident-initiated iodine spike dose. The methodology assumes that the RCS leakage is occurring at the Technical Specification limits. Historically, nuclear power plants operate at an RCS leakage much less than the Technical Specification limits. Units do not operate at the extreme limit of the analytical operating condition boundaries for all of the parameters used in the concurrent iodine spike dose calculations. Thus, operations under actual operating conditions would result in lower concurrent iodine spike doses than those calculated in the analyses.

The design basis accident concurrent iodine spike analysis typically follows the NRC Standard Review Plan model (15.1.15, App. A, III.4.b) which specify that the iodine release rate from the fuel rods to the primary coolant be increased to a value 500 times greater than the pre-event steady state iodine release rate as a result of the DBA event occurring. This 500-factor increase is believed to be a very conservative assumption for what may occur to the iodine release rate during a design basis accident.

Since the RCS iodine activity is typically not near the limit and the dose analysis model is very conservative, the non conservatism identified in the dose analysis do not represent a substantial safety hazard.

NRC AWARENESS/REPORTABILITY CONSIDERATIONS

The nonconservatisms in the accident initiated iodine spiking rate dose analysis was reported by an operating plant to the NRC. See BVPS Unit 1 LER #: 99-002-00 (NRC Part 21 Log 99-15-1) for the complete text.

RECOMMENDED ACTIONS

To assure a conservative determination of the iodine appearance rates, these rates should be recalculated considering all of the above items. In summary these are:

- Actual letdown flow at which the plant is operating or intends to operate for extended periods plus uncertainties
- Identified primary coolant leak rate of 10 gpm (typical) including a primary to secondary leak rate of 1.0 gpm (typical)
- Unidentified primary coolant leak rate of 1.0 gpm (typical)
- Removal of all iodine from the letdown stream by the demineralizer (i.e., infinite DF)

Accident dose analyses that utilize the accident-initiated iodine spike should be recalculated to incorporate the conservatively determined iodine spiking rates. The recalculation of accident doses may include modifications to the accident analysis assumptions that would reduce the impact of an increase in the appearance rate. For example, if the thyroid doses were calculated using the dose conversion factors from TID-14844, the factors from ICRP Publication 30 could be used for a benefit. In addition, if the existing analyses assume an iodine partition coefficient of only 0.1 in the intact steam generators, the value could be reduced to 0.01 consistent with the NRC's Standard Review Plan.

Until reanalysis of accident doses are performed, existing analyses can be demonstrated to remain bounding by limiting plant operation to an appropriate fraction of the Technical Specification limit on primary coolant iodine activity as discussed in the following section. Most plants have a Technical Specification limit of 1.0 μCi/g of Dose Equivalent I-131 but operate with levels far below this limit. Other options discussed previously including limiting letdown flow may also be applied.

The following estimated reductions in the primary coolant iodine activity limit would be required to assure that the accident analyses remain representative of potential operation (specific plant values would require an individual determination). These reductions are based on assuming that all of the identified nonconservatisms were used in the existing determination of iodine appearance rates. These reductions in coolant equilibrium activity are provided below in terms of the letdown purification flow rates used at the plants. These examples assume a 10 percent uncertainty on the letdown flow and leakage at the limiting values.

The first sets of numbers are for "normal" letdown flow meaning one orifice in use.

"Normal" letdown flow and a Technical Specification limit of 1.0 μCi/g Dose Equivalent I-131

	Letdown Flow Flow (gpm)	Correction Factor	Revised Iodine Limit (μCi/g)
Two-Loop plants	40	1.56	0.64
Three-Loop plants	60	1.44	0.69
Four-Loop plants	75	1.40	0.71
4XL plants	100	1.36	0.73

For the plants that have modified operation to increase the normal letdown flow to the "maximum purification" flow rate using two letdown orifices, the following recommendations apply.

"Maximum Purification" letdown flow and a Technical Specification limit of 1.0 μCi/g Dose Equivalent I-131

	Letdown Flow Flow (gpm)	Correction Factor	Revised Iodine Limit (μCi/g)
Two-Loop plants	80	2.78	0.36
Three-Loop plants	120	2.67	0.37
Four-Loop plants	120	2.13	0.47
4XL plants	198	2.55	0.39

These estimates are provided as examples. Specific plant values require individual determination and verification of plant specific parameters and inputs.

REFERENCES:

BVPS Unit 1 LER #: 99-002-00, 03/03/99, Non-Conservative Concurrent Iodine Spike Radiological Dose Calculation Methodology, (NRC Part 21 Log 99-15-1), http://www.nrc.gov/NRC/PUBLIC/PART21/1999/1999151.html.

TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," 3/23/62, AEC Division of Licensing and Regulation

ICRP PUBLICATION 30, SUPPLEMENT TO PART 1, VOLUME 3, NO. 1-4, "LIMITS FOR INTAKES OF RADIONUCLIDES BY WORKERS," 1979