ENERGY NORTHWEST People·Vision·Solutions	CALCULATION COVER SHEET ORIGINAL	BDC/PDC PDC 2406

Equipment Piece No.	Project Columbia	Page 1.0 / Cont'd on Page
	Discipline: Nuclear	Calculation No. NE-02-04-06
		Quality Class 1
	Remarks	

TITLE/SUBJECT/PURPOSE

Main Steamline Break Accident Off-site and Control Room Doses

Purpose

The purpose of this calculation is three-fold:

(1) To perform a QA evaluation to determine the local (i.e., on-site) atmospheric dispersion factor for the superheated "puff" produced by the steam released from the Columbia Generating Station main steam line break (MSLB) design basis accident (DBA),

(2) To apply this dispersion factor to the dose analysis using Polestar's STARDOSE code to calculate Control Room (CR) doses, and

(3) To perform off-site dose calculations.

CALCULATION REVISION RECORD

REVISION NO.	STATUS/ F,P, OR S	REVISION DESCRIPTION	INITIATING DOCUMENTS	TRANSMITTAL NO.
0	F	New Calculation		
1	F	Add Appendix D " RADTRAD ANALYSIS"		19196

PERFORMANCE VERIFICATION RECORD

REVISION NO.	PERFORMED BY/DATE	VERIFIED BY/DATE	APPROVED BY/DATE
0	Jim Metcalf	Bernard Nowack	Bernard Nowack
1	Mohammed Abu-Shehadeh	Linda Woosley	Shaw Bian
	Mohammed AbuShehadeh 9/28/04	LBWoosley 9/30/04	SABian 9/30/04

* Study calculations shall be used only for the purpose of evaluating alternate design options or assisting the engineer in performing assessments.

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		Revision No. 1	

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		Revision No. 1	

Calculation/CMR NE-02-04-06 **Revision** 1
was verified using the following methods:

☒ Checklist Below ☐ Alternate Calculation(s)

Checklist Item | **Verifier Initials**

Clear statement of purpose of analysis .... LSW

Methodology is clearly stated, sufficiently detailed, and appropriate for the proposed application .... LSW

Does the analysis/calculation methodology (including criteria and assumptions) differ from that described in the Plant or ISFSI FSAR or NRC Safety Evaluation Report, or are the results of the analysis/calculation as described in the Plant or ISFSI FSAR or NRC Safety Evaluation Report affected?
☒ Yes ☐ No .... LSW
If Yes, ensure that the requirements of 10 CFR 50.59 and/or 10 CFR 72.48 have been processed in accordance with SWP-LIC-02. .... LSW

Does the analysis/calculation result require revising any existing output interface document as identified in DES-4-1, Attachment 7.3?
☐ Yes ☒ No .... LSW
If Yes, ensure that the appropriate actions are taken to revise the output interface documents per DES-4-1, section 3.1.8 (i.e., document change is initiated in accordance with applicable procedures). .... LSW

Logical consistency of analysis .... LSW
- Completeness of documenting references .... LSW
- Completeness of input .... LSW
- Accuracy of input data .... LSW
- Consistency of input data with approved criteria .... LSW
- Completeness in stating assumptions .... LSW
- Validity of assumptions .... LSW
- Calculation sufficiently detailed .... LSW
- Arithmetical accuracy .... LSW
- Physical units specified and correctly used .... LSW
- Reasonableness of output conclusion .... LSW

Supervisor independency check (if acting as Verifier) .... LSW - N/A
- Did not specify analysis approach
- Did not rule out specific analysis options
- Did not establish analysis inputs .... LSW - N/A

If a computer program was used: .... LSW
- Is the program appropriate for the proposed application?
- Have the program error notices been reviewed to determine if they pose any limitations for this application?
- Is the program name, revision number, and date of run inscribed on the output?
- Is the program identified on the Calculation Method Form? If so, is it listed in Chapter 10 of the Engineering Standards Manual? .... LSW

Other elements considered:
NE-02-02-17 RADTRAD V&V .... LSW

If separate Verifiers were used for validating these functions or a portion of these functions, each sign and initial below.
Based on the foregoing, the Calculation/CMR is adequate for the purpose intended.

Verifier Signature(s)/Date | Verifier Initials
LS Woosley [signature] 9/30/04 | LSW

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		Revision No. 0	

NO	AUTHOR	ISSUE DATE/ EDITION OR REV.	TITLE	DOCUMENT NO.
1	K. Eckerman et al, Oak Ridge National Laboratory, Oak Ridge, TN	1988	"Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", Federal Guidance Report No. 11, page 136	EPA-520/1-88-020
2	U.S. Nuclear Regulatory Commission	July, 2000	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	Regulatory Guide 1.183
3	Polestar Applied Technology, Inc.	Revision 1	Project QA Plan for Application of Alternate Source Term to Non-LOCA DBAs for Columbia Generating Station	PSAT 206CT.QA.02.01
4	Polestar Applied Technology, Inc.	Revision 2	Implementing Procedure for Application of the Alternate Source Term to LOCA and Non-LOCA DBAs for Energy Northwest Columbia Generating Station	PSAT 206CT.QA.01.02
5	U.S. Nuclear Regulatory Commission	June, 2003	Atmospheric Relative Concentrations For Control Room Radiological Habitability Assessments At Nuclear Power Plants	Regulatory Guide 1.194
6	Polestar Applied Technology, Inc.	Revision 0	STARDOSE Validation Report	PSAT CI09.05
7	Polestar Applied Technology, Inc.	Revision 0	STARDOSE Users Manual	PSAT CI09.06
8	Energy Northwest		Tech. Spec. 3.4.8 "RCS Specific Activity"	TS Amendment No. 169
9	Energy Northwest	Amendment 53, Nov. 1998	Columbia Generating Station Final Safety Analysis Report	WNP-2 FSAR
10	Energy Northwest	Rev. 8, Sept. 1997	Columbia Generating Station, WNP-2 Systems Data Sheet	82-RSY-0300-T3, SC
11	Polestar Applied Technology, Inc.	Revision 0	Dose Calculation Data Base	NE-02-04-1
12	Humphreys, S.L., et al.	December, 1997	RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation	NUREG/CR-6604

ENERGY NORTHWEST People·Vision·Solutions	CALCULATION OUTPUT INTERFACE DOCUMENT REVISION INDEX	Page 1.4	Cont'd on
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Prepared by/Date: [signature] 6/17/04	Verified by/Date: BLY 6-18-04	Revision No. 0	

The below listed output interface calculations and/or documents are impacted by the current revision of the subject calculation. The listed output interfaces require revision as a result of this calculation. The documents have been revised, or the revision deferred with Manager approval, as indicated below.

AFFECTED DOCUMENT NO.	CHANGED BY (e.g., BDC, SCN, CMR, Rev.)	CHANGED DEFERRED (e.g., RFTS, LETTER NO.)	DEPT. MANAGER *
FSAR 15.6.4	PDC 2406		

* Required for deferred changes only.

Discussion of Results | Revision No. 0 | REV. BAR

Control Room Dose: The Control Room (CR) dose was analyzed for two release scenarios: the first is the direct release of steam to the atmosphere via the blowout panels (this will be the license basis case), the second is the release via the Turbine Generator Building (TGB) to the atmosphere. No credit was taken for the remote intakes or the Control Room Emergency Filtration (CREF) system. The dose conversion factors (DCFs) are based on Reference 1 which is recommended by Reference 2. The dose for each release scenario was calculated with and without the effect of iodine spiking. A summary of the dose results of these two scenarios is as follows:

Control Room Dose From Direct Release to the Atmosphere

	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
Dose without iodine spiking (rem)	2.83E-05	8.95E-02	8.95E-02	5
Dose with iodine spiking (rem)	5.65E-04	1.79	1.79	5

Control Room Dose, Release Via TGB, Mixing w/ TGB Air

	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
Dose without iodine spiking (rem)	1.27E-05	4.03E-02	4.03E-02	5
Dose with iodine spiking (rem)	2.54E-04	0.81	0.81	5

Off-site Dose: The off-site (EAB and LPZ) doses (including iodine spike) were calculated using the formula given below. Per RG 1.183 (Reference 2), the MSLB dose limits with and without iodine spiking are 25 and 2.5 rem, respectively. Since the source from iodine spiking is 20 times higher than that from equilibrium iodine (4 µCi/g vs. 0.2 µCi/g), the dose corresponding to the spike is 20 times higher than that corresponding to equilibrium iodine. Therefore, the spike dose is more limiting since it results in a higher percentage of the dose limit. The results are summarized in the following table.

$$\text{Dose (rem)} = [\text{Activity Release (Ci)}] \times [\chi/Q\ (s/m^3)] \times [\text{Breathing Rate } (m^3/s)] \times [\text{DCF (rem/Ci)}]$$

Off-site Doses (rem) with and without Iodine Spike

With Iodine Spike	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
EAB Dose (rem)	Negligible	0.398	0.398	25
LPZ Dose (rem)	Negligible	0.109	0.109	25

Without Iodine Spike	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
EAB Dose (rem)	Negligible	0.020	0.020	2.5
LPZ Dose (rem)	Negligible	0.0055	0.0055	2.5

Conclusions

CR Dose Results - The conclusion from these results is that the MSLB CR limiting dose of 1.79 rem (corresponding to direct release to the environment with iodine spiking) is well below the 5.0 rem TEDE regulatory limit for control room operator exposure given in Reference 2 for BWR MSLB.

Off-site Doses Results - The conclusion from these results is that the MSLB off-site doses (including iodine spike) of 0.398 rem and 0.109 rem for EAB and LPZ, respectively, are well below the 25 rem TEDE regulatory limit from Reference 2 for BWR MSLB with spiking. In fact, these results are less than the 2.5 rem TEDE regulatory limit for BWR MSLB without spiking considered.

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Prepared by / Date: 6/17/04	Verified by/Date: BLV 6-18-04	Revision No. 0

Analysis Method (Check appropriate boxes)

☒ Manual (As required, document source of equations in Reference List)

☒ Computer ☐ Main Frame ☐ Personal

☐ In-House Program

☐ Computer Service Bureau Program

☐ BCS ☐ CDC ☐ PCC ☐ OTHER

☒ Verified Program: Code name/Revision STARDOSE, version 1.01

☐ Unverified Program: Document in Appendix B

Approach/Methodology REV. BAR

The methodology and scope of this calculation is consistent with References 3 and 4.

Calculation of CR Dispersion Factor (χ/Q) - The Instantaneous Puff Release model described in Reference 5 to determine the time-dependent dispersion of a non-rising, ground level, instantaneous puff release was used to calculate the CR χ/Q value. The initial volume of the puff is established by the amount of steam released by the MSLB and by the flashing of entrained liquid. The calculation of this initial steam volume (and the DE I-131 concentration) is the first step of the calculation.

The puff centerline is then assumed to pass directly over the local CR air intake. The release point from the TGB is assumed to be far enough away from the normal CR air intake to permit the puff to be fully extended (i.e., $x \sim 3\sigma$ for the puff) before movement across the CR air intake begins. This maximizes the time-integrated, normalized concentration (expressed in sec/m^3). No credit is taken for vertical (z-direction) expansion in performing the normalized concentration integration.

Parameters for the MSLB DBA include the mass of liquid-steam mixture released, the timing of release, the temperature of the liquid-steam mixture, and the iodine concentration in the release. These parameters are used to obtain the initial conditions of the released steam puff. The Reference 5 methodology then establishes the puff's transit time, the normalized concentration as a function of distance traveled in the downwind or "x" direction, and, finally, the time-integrated, normalized centerline concentration.

CR Dose - For the licensing-basis CR dose calculation, the transport pathway is based upon direct release to the environment. For completeness, a second transport pathway via the TGB is also considered (see Assumptions A-7 and A-8 for further discussion). The STARDOSE computer code [references 6 and 7] is used to determine the CR dose. A STARDOSE LIBFILE1.TXT file was created with the Dose Equivalent (DE) I-131 inventory for the Columbia MSLB using the DE I-131 coolant concentration from Reference 8 with consideration, also, of the potential for iodine spiking. A STARDOSE INPUT.DAT file was also prepared to represent exposure to the CR operator equivalent to that provided by the passing puff. This equivalency is provided by introducing into the CR at the start of the dose calculation the proper fraction of the total DE I-131 release, and then purging that DE I-131 from the CR at the normal flow rate.

Off-site Dose - For the licensing-basis off-site dose calculation, the same DE I-131 source term as that used for the CR dose calculation is used. It is conservatively assumed that the only transport pathway is a direct release to the environment. The plume dilution effect due to buoyant rise is also conservatively neglected (see Assumption C-2 for further discussion). Because of the simplicity of the off-site dose model, a manual calculation is employed.

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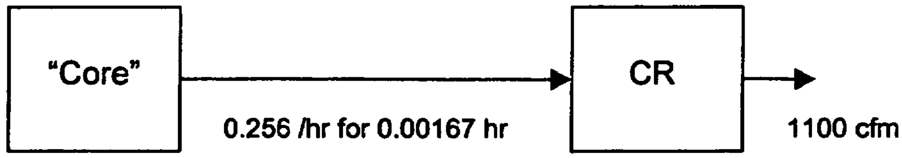

Figure 1. STARDOSE Model for Fission Product Release

Purpose: The purpose of this calculation is to perform radiation dose calculations following a MSLB accident at Columbia. The dose calculations will include the control room, the EAB, and the LPZ.

These calculations are being performed in accordance with References 3 and 4. The activity released in the MSLB accident is from fission products dissolved in the coolant. This activity is limited by Technical Specification to 0.2 μCi/g Dose-Equivalent (DE) I-131 with a short-term allowance for iodine spiking to 4.0 μCi/g DE I-131. The activity concentration (Ci/m^3) to which the CR is exposed (and which must be applied to the dose calculation) is reduced as the plume is diluted by entrained air and expansion of the puff. Although the puff would be buoyant, no credit is taken for buoyant rise. Table 1 presents the input parameters used in the calculations.

Table 1. Design Input Parameters

Columbia Design Input Parameter	Parameter Value	Basis
Maximum time for MSIV closure	6 sec	Reference 11
Approx. volume of TGB	5.71E6 ft^3	Reference 11
Liquid release from MSLB	105,000 lbm	Reference 11
Steam release from MSLB	25,000 lbm	Reference 11
RCS pressure	1060 psia (552 F)	Reference 11
Blowout panel locations for MSLB	Panels A to TGB (N. end of tunnel) and D direct to environment (via B and C) (E. end of tunnel)	Reference 11
Distance from MSLB release point to normal CR intake for Panel D	240 ft = 73 m	Reference 11
Distance from MSLB release point to normal CR intake for Panel A (via TGB)	200 ft = 61 m	Reference 11
Plume transit velocity	1 m/s	Reference 11
Coolant iodine inventories	0.2 μCi/g DE I-131	Reference 11
Iodine spiking factor	20 (increasing coolant activity to 4 μCi/g DE I-131)	Reference 11
Radioactivity release rate to environment	Instantaneous	Reference 5
Vol. of CR	214,000 ft^3	Reference 11
CR occupancy factor	1	Reference 11
CR normal, unfiltered makeup flow	1100 cfm	Reference 11
CR Breathing Rate	3.5E-4	Reference 11
Chi/Q, EAB	1.81E-4 sec/m^3	Reference 11
Chi/Q, LPZ	4.95E-5 sec/m^3	Reference 11
Dose Conversion Factor for I-131 CEDE	32893 rem/Ci	References 1,12

The various transport pathways, geometries, and puff/plume dilutions being considered in the main body of this calculation are summarized in Table 2 below:

Table 2. Summary of Cases

Dose Calculation	Transport Pathway	Geometry	Plume Dilution
CR	Direct to environment (primary case – used for licensing basis)	Instantaneous puff release – Gaussian distribution	Air entrainment and expansion during transit
CR	Via TGB (secondary case for information – not for licensing basis)	Instantaneous puff release – Gaussian distribution	Pre-dilution in TGB - air entrainment and expansion during transit
Off-site	Direct to environment	Plume*	Air entrainment and expansion during transit

* The PAVAN code was used to calculate the off-site χ/Q values.

Calculations of Radiation Doses: The calculation of radiation doses from a MSLB is divided into two main categories, the control room dose and the off-site doses. The details of the calculation are given below:

I. Calculation of Control Room Doses: Calculations of the control room doses involve the calculation of the effective puff relative concentration (X/Q), the source term (i.e; amount of activity released), and the development of a STARDOSE model to calculate the radiation dose. In order to facilitate an acceptable calculation for the effective puff relative concentration (X/Q), the following assumptions and their justifications are introduced.

<u>Assumption A-1</u> The main steam isolation valve (MSIV) closure time will not exceed 6 seconds. The release of steam resulting from the MSLB (through blowout panels in the steam tunnel) is assumed to be instantaneous. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount passing through the MSIVs prior to closure.

<u>Justification</u> The Columbia FSAR (Section 15.6.4.4, page 15.6-7 of Reference 9) states that the MSLB steam and liquid discharge is based on MSIVs closing in 6 seconds. Since this is the basis for the current MSLB radiological licensing evaluation, it is reasonable to assume that this is the maximum allowed closure time. This time duration is small compared to the exposure time of interest for the CR, and in any event it is conservative to assume instantaneous release.

The FSAR evaluation of mass released in the MSLB is based on the 6 second closure time and states that a steam-water mixture flows from the break until the MSIV has closed.

<u>Assumption A-2</u> The puff from the liquid-steam release (including the flashed steam) is released at ground-level with an initial volume corresponding to standard atmospheric conditions. Activity within the puff becomes normally distributed by dilution with air. No buoyancy is considered. The liquid (assumed to contain no residual activity – a very conservative assumption) settles by gravity.

<u>Justification</u> Since the iodine is in solution, and nearly all of it will tend to stay with the liquid, it is conservative to assume that all of the iodine activity partitions with the steam and becomes airborne. For the most part, the released liquid consists of large droplets from the blowdown that will settle quickly without complete evaporation.

The puff is allowed to entrain air and to expand slightly as it moves downwind. According to the Reference 5 model, the integration of the normalized puff activity concentration as it crosses the CR air intake is performed from x to $+3\sigma$ (where x is the distance from the release point to the receptor; i.e., the air intake, and "σ" includes the increase in σ by expansion in the downwind direction over the distance x).

<u>Assumption A-3</u> Control room ventilation remains in normal mode. The normal air intake is the one used for analyzing dispersion.

<u>Justification</u> For the analysis, there is no "FAZ" signal credited to start emergency control room ventilation. No credit is taken for operator actions. MSIV isolation actuates on high steam flow.

<u>Assumption A-4</u> Control room ventilation intake flow is unfiltered. No consideration is needed of unfiltered inleakage in calculating dispersion.

<u>Justification</u> For the analysis ventilation remains in normal mode. No credit is taken for filtration at all.

<u>Assumption A-5</u> The time required for the plume to transit to the CR air intake is based on the plume moving with a horizontal velocity of 1 m/s. The local air intake is used as the basis for the transit time.

<u>Justification</u> There are three CR intakes at Columbia. Two of these are remote, located away from the power block (400 feet or more from the MSLB release location described in Assumption A-7). The third is the local intake which is contiguous with the CR building (see Appendix A of Reference 11). This is the CR intake closest to the MSLB release location; and thus, it is assumed that the plume translates directly to the local CR intake.

REV. BAR

Assumption A-6 The steam-air mixture may be treated as a perfect gas.

Justification The perfect gas assumption is applicable to low pressure, high temperature gases where there are minimal interaction forces between gas molecules. The puff is at atmospheric pressure and high temperature.

Assumption A-7 As noted in the CR Dose portion of the Methodology section, the primary release location (transport pathway), and that upon which the final, licensing-basis results are based, is direct release to the environment. Release via the TGB (with brief confinement in the TGB) has also been considered.

Justification Release directly to the environment is consistent with RG 1.183 which states that for MSLB, all radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously. This is also consistent with what would be expected for the MSLB at Columbia based on the following.

Per Reference 10 and Reference 11, Appendix A, with the MSLB in the steam tunnel or tunnel extension (located in the TGB), there are three blowout panels that are designed to vent to prevent overpressurization. Panel A at the north end of the tunnel vents into the TGB. Panels B and C vent to a vent-way which vents directly to the atmosphere via blow-out panel D. Thus, the release will vent either directly to the atmosphere or into the TGB.

If the release occurs into the TGB, approximately 65,000 lbm of hot steam will appear essentially instantaneously into a volume of ~5.7E6 ft3 (Reference 11). Based on the perfect gas law, such a steam release would lead to a pressure increase of ~5.4 psi within the TGB (based on an unmixed, isothermal compression such that each component experiences the same fractional reduction in volume but no increase in temperature – see Attachment 1). Thus for the TGB to remain intact, it would have to withstand a pressure of ~5.4 psig. (Per Attachment 1, even a well-mixed model results in a pressure of ~3.7 psig). A fraction of this pressure (~1 psig) would be expected to catastrophically fail the TGB siding; thus, the release into the TGB can be treated as a release directly to the environment.

Release via the TGB with brief confinement has also been considered and is discussed further in Assumption A-8. This is not, however, considered appropriate for the licensing basis and is included only for completeness.

Assumption A-8 For release into the TGB, there are two possibilities to consider. One is that the TGB fails such that it, in effect, provides no confinement of the steam puff and the result is similar to a release directly to the environment. This is the primary release path and is addressed by the release directly to the atmosphere discussed in Assumption A-7. The other case is where the TGB tends to briefly confine the puff, with release from the TGB at one or more specific failure locations. In this latter case, it is assumed that the steam puff mixes with the air in the TGB prior to release from the TGB.

Justification The TGB is about 5.7E6 ft^3 in net free volume. Assuming for the moment that the TGB remains largely intact and tends to briefly confine the steam release as it is vented through blowout Panel A, the puff will mix rapidly with the air in the TGB by jet entrainment and density driven exchange. Thus the release to the atmosphere from the TGB will be a mixture of air and steam. To account for isolated volumes in the TGB and displacement of TGB air at the time of venting into the TGB, the mixing is assumed to involve only 2/3 of the gross building volume. The 5.7E6 ft^3 is the net building volume allowing 20% for equipment and internal structures. Therefore, 2/3 of the gross volume is 0.67*5.7E6*1.25 = 84% of the net TGB volume of 5.7E6 ft^3; i.e., the mixture released from the TGB is the steam volume plus approximately 84% of the TGB volume as air.

I.1 Calculation of CR Effective Puff Relative Concentration (χ/Q): The Instantaneous Puff Release model described in Reference 5 to determine the time-dependent dispersion of a non-rising, ground level, instantaneous puff release was used to calculate the control room χ/Q value. The details of calculating this relative concentration are given in the following steps:

I.1.1 Calculation of Initial Steam Volume: The initial volume of the puff is established by the amount of steam released by the MSLB and by the flashing of entrained liquid. The calculation of this initial steam volume (and the DE I-131 concentration) is the first step of the calculation.

I.1.1.1 Evaluation of the Initial Conditions of Steam Release: Per Reference 9, the liquid-steam mixture is released over a period of ~6 seconds (assumed to be instantaneous per Assumption 1), 105,000 lbm as liquid and 25,000 lbm as steam, and the RCS pressure is 1060 psia. The temperature of the liquid-steam mixture at the time of the release to ambient is the saturation temperature corresponding to 1060 psia which is 552 F. Since the liquid is superheated at ambient pressure, some of this liquid will flash to steam. Per Reference 5 and Assumption 2, the steam will form the puff, and the unflashed liquid will settle by gravity.

I.1.1.2 Determination of the liquid flashing fraction (ff): The fraction of the released liquid that flashes into steam can be determined from the following energy balance equation:

$mh = m_g h_g + m_l h_l$

where: m = initial liquid mass (lbm)
h = initial liquid enthalpy (Btu/lbm)
m_g = flashed steam mass (lbm)
h_g = flashed steam enthalpy (Btu/lbm)
m_l = unflashed liquid mass (lbm)
h_l = unflashed liquid enthalpy (Btu/lbm)

and the unflashed liquid and flashed steam are at atmospheric pressure and saturation temperature corresponding to atmospheric pressure (212 F).

The flashing fraction, ff, is

$$ff = m_g/m = (mh - m_l h_l)/m/h_g$$

$$= (h - m_l h_l/m)/h_g$$

Since

$$m_l/m = (m - m_g)/m = 1\text{-}ff$$

we have

$$ff = (h - (1\text{-}ff)h_l)/h_g$$

Thus,

$$ff = (h - h_l)/(h_g\text{-}h_l)$$

Using the steam tables,

$$h(552\text{ F}) = 552\text{ Btu/lbm}$$

$$h_l(212\text{ F}) = 180.2\text{ Btu/lb}$$

$$h_g(212\text{ F}) = 1150.5\text{ Btu/lb}$$

Thus,

$$ff = (552 - 180.2)/(1150.5 - 180.2)$$

$$= 0.383$$

This means that 38.3% of the released liquid flashes into steam in addition to the 25,000 lbm of coolant initially released as steam.

I.1.1.3 Mass of Liquid Flashed and Total Steam Mass: the mass of flashed steam is calculated by multiplying the flashing fraction by the amount of the released liquid:

$$m_g = 0.383 \times 105{,}000 = 4.02E4 \text{ lbm.}$$

The total initial steam puff mass is the sum of the initial mass of released steam (25,000 lbm) and the mass of steam flashed from the liquid (4.02E4 lbm):

The total steam mass (lbm) = 25,000 lbm + 4.02E4 lbm = 6.52E4 lbm

I.1.1.4 Volume of Steam: The initial volume of the steam puff is calculated as follows:

The weighted average temperature of the mixture of the steam released at 552 F and the steam flashed from the liquid at 212 F is:

$$T_b = (4.02E4 \times 212 + 25000 \times 552)/6.52E4 = 342 \text{ F} = 802 \text{ R}$$

Therefore, the initial volume of the plume (pure steam) is:

$$V = 6.52E4 \text{ lbm}/\rho_s$$

where $\rho_s = 0.0311$ lbm/ft^3 (super heated steam density at 802 R). Thus,

$$\text{Initial puff volume} = V = 2.1E6 \text{ ft}^3 = 5.95E4 \text{ m}^3 \qquad (1)$$

I.1.2 Calculation of the Effective Puff Relative Concentration:

The instantaneous Puff Release model described in RG 1.194 (Reference 5) has been used to evaluate the relative concentration (χ/Q). The activity release to the environment must occur over a period of no longer than about one minute for a release to qualify as a puff release. The diffusion equation for an instantaneous puff ground level release, with no puff rise and no crosswind offset (i.e., the center of puff is assumed to pass directly over control room intake), integrated over the duration of the puff passage is:

$$\frac{\chi}{Q}(x,u,k,h) = \frac{\displaystyle\int_0^T \frac{2}{(\sigma_z^2(x,k)+\sigma_l^2)^{\frac{1}{2}}(2\pi)^{\frac{3}{2}}(\sigma_{x,y}^2(x,k)+\sigma_l^2)}.\exp\left[-\frac{1}{2}\left(\frac{(x-u.t)^2}{(\sigma_{x,y}^2(x,k)+\sigma_l^2)}+\frac{h^2}{(\sigma_z^2(x,k)+\sigma_l^2)}\right)\right].F(t)dt}{\displaystyle\int_0^T F(t)dt} \qquad (2)$$

where:

- χ = Integrated concentration at control room intake, Ci-m^3/sec
- Q = Release quantity, for nuclide, Ci
- x = Release point to receptor distance, m. The distance from the blowout panels to the local CR intake is 200 ft (~ 61 m).
- u = Wind speed, m/sec. Per RG 1.194 the wind speed is assumed to be 1 m/sec
- k = Stability Class. Per RG 1.194 stability class F will be used.
- h = Difference in elevation between the physical release point and the control room intake, m. If the control room intakes is at a higher elevation than the release point and the puff is buoyant, assume h = 0.
- T = Transit time - time for trailing edge of puff to pass control room intake, sec. Per RG 1.194, the following formula should be used to calculate the transient time:

$$T = \frac{x + 3.\left[(\sigma_{x,y}(x,k) + \sigma_l)\right]}{u} \qquad (3)$$

F = Control room total intake flow rate, cfm. (If the control room intake flow rate is constant over the period 0 to T seconds, the F(t) terms can be omitted), since at Columbia the CR intake flow rate is not a function of time, F(t) in the numerator and denominator will cancel each other.

$\sigma_{x,y}(x,k)$ = Standard deviation, m, of the puff in the horizontal along the wind direction and cross-wind directions at the receptor location. Figure 4 in Reference 5 is used with the distance to the receptor and the stability class to determine $\sigma_{x,y}$ at the receptor.

For a distance x = 61 m and stability class F, the value of $\sigma_{x,y}(x,k)$, obtained from Figure 4 in Reference 5, is:

$\sigma_{x,y}(x,k) = 2.9$ m.

$\sigma_z(x,k)$ = Standard deviation, m, of the puff in the vertical cross-wind direction at the receptor location. Figure 5 in Reference 5 may be used with the distance to the receptor and the stability class to determine σ_z at the receptor; but in this case, expansion in the z direction is conservatively neglected.

σ_I = Per RG 1.194, the initial standard deviation, m, is given by the following formula:

$$\sigma_I = \left[\frac{2.V}{(2\pi)^{\frac{3}{2}}}\right]^{\frac{1}{3}} \qquad (4)$$

But per equation (1) above, V=5.95E4 m^3, hence,

$\sigma_I = 19.62$ m

The transient time, T, can now be calculated by introducing the values of x, u, σ_I, and $\sigma_{x,y}(x,k)$ into equation (3), the result is:

T = 128.6 seconds.

In order to simplify the integral in equation (2) above, the following constants are defined:

$$A = \frac{2}{(\sigma_z^2(x,k)+\sigma_I^2)^{\frac{1}{2}}(2\pi)^{\frac{3}{2}}(\sigma_{x,y}^2(x,k)+\sigma_I^2)} = 1.65\text{E-5 m}^{-3} \qquad (5)$$

$$B = \frac{\frac{1}{2}}{(\sigma_{x,y}^2(x,k)+\sigma_I^2)} = 1.27\text{E-3 m}^{-2} \qquad (6)$$

$$C = X \qquad (7)$$

$$D = u \qquad (8)$$

Introducing the definitions in equations (5) through (8) into equation (2) and multiplying the right-hand side of the equation by $\left(\frac{\sqrt{\pi}}{\sqrt{\pi}}.\frac{2}{2}\right)$, the equation becomes:

$$\chi / Q = \left(\frac{A\sqrt{\pi}}{2}\right)\frac{2}{\sqrt{\pi}}\int_0^T e^{-B(C-Dt)^2}\,dt \qquad (9)$$

Furthermore, we need to introduce the following definitions:

$$E = B^{1/2} = 0.0356\ \mathrm{m}^{-1} \qquad (10)$$

$$F = EC = 2.17 \qquad (11)$$

$$G = ED = 0.0356\ \mathrm{sec}^{-1} \qquad (12)$$

$$\nu = F - Gt \qquad (13)$$

Introducing equations (10) through (13) into equation (9) yields the following equation:

$$\chi / Q = \left(\frac{A\sqrt{\pi}}{-2G}\right)\frac{2}{\sqrt{\pi}}\int_0^T e^{-\nu^2}\,d\nu \qquad (14)$$

Equation (14) represents the definition of the error function, (erf), which has the following solution:

$$\chi / Q = \left(\frac{A\sqrt{\pi}}{-2G}\right)\frac{2}{\sqrt{\pi}}\int_0^T e^{-\nu^2}\,d\nu = \left(\frac{A\sqrt{\pi}}{-2G}\right)[erf(\nu(T)) - erf(\nu(0))] \qquad (15)$$

where $erf(\nu) = (2/\pi^{1/2})\Sigma\{(-1)^k \nu^{(2k+1)}/[k!(2k+1)]\}$ summed from k = 0 to infinity. Tables of the error function are readily available.

For t = T = 128.6 sec, $erf(\nu)$ = erf(F-GT) = erf(F-G(128.6)) = erf(2.17-(0.0356)(128.6)) = erf(2.17-4.58) = erf(-2.41)
For t = 0 sec, $erf(\nu)$ = erf(F-Gt) = erf(F-G(0)) = erf(2.17).

Tables for $erf(\nu)$ typically extend to $\nu = 2$ which give erf(2) = 0.995322 by using the first 18 terms of the error function expansion, (it has the value of 1.0 at infinity). Therefore, in order to obtain accurate values of the error function at $\nu = -2.41$ and $\nu = 2.17$, the first 25 terms of the series function were used, therefore:

erf(-2.41) = -0.999346 and erf(2.17) = 0.997851

The term $[erf(\nu(T))-erf(\nu(0))]$ in equation (15) can now be evaluated as follows

$$[erf(\nu(T))-erf(\nu(0))] = [(-0.999346)-(0.997851)] = -1.997 \qquad (16)$$

The second part of equation (15) is evaluated as follows:

$$-A\pi^{1/2}/2G = -(1.65\mathrm{E}{-5})(1.77)/2/0.0356 = -4.10\mathrm{E}{-4}\ \mathrm{sec/m^3}. \qquad (17)$$

Multiplying equation (16) by (17) gives the X/Q value:

$$\chi/Q = (-4.10\mathrm{E}{-4})(-1.997) = 8.19\mathrm{E}{-04}\ \mathrm{sec/m^3}$$

Multiplying this value by the intake flowrate of 1100 cfm = 0.520 m^3/sec, one obtains the fraction of the release, F_{CR}, that enters the CR. This fraction is:

In the above analysis the assumption was made that the steam is directly released to the atmosphere via the blowout panel D. If one assumes that the steam release mixes with 2/3 of the TGB gross volume prior to being released to the environment, the puff will become larger and more dilute. Although the puff is more dilute, the exposure time will also become greater. It is expected that the dilution (which increases with the volume of the puff) will have a more significant effect than the extended exposure time (which increases only with the linear dimension of the puff). Therefore, one would expect the fraction of the release which enters the CR to decrease. The following evaluation confirms that expectation. Assume that brief holdup and mixing occurs in the TGB as noted in Assumption A-8; i.e., assume that the vented steam mixes with the TGB air prior to release from the TGB itself. Thus, the release via the TGB is pre-diluted (see "Methodology"). As noted in the Methodology section and discussed in Assumption A-7, this transport pathway evaluation is provided for completeness and is not considered appropriate for use in the licensing basis.

The puff volume was calculated as follows. Per Assumption A-8, the steam mixes with 84% of the air in the TGB volume; i.e., 5.95E+04 m^3 (steam) + 0.84 * 5.7E+06 ft^3 (TGB volume) / 35.3 ft^3/m^3 = 5.95E+04 m^3 + 1.36E+05 m^3 = 1.96E+05 m^3.

Applying this volume to equation (4), the σ_1 increases by the ratio $(1.96E+05/5.95E+04)^{1/3}$ = 1.49. The corresponding values of A and B become (from equations (5) and (6)) 5.03E-06 m^{-3} and 5.79E-04 m^{-2}, respectively. Observing that equation (16) is already -1.997 and cannot be greater than -2 (absolute value), it is evident that the χ/Q for the greater volume will vary as A/G from equation (15). Since G = ED (equation (12)) and E = $B^{1/2}$ (equation (10)), and since D = u (equation (8)) = 1.0 m/sec, A/G = 2.09E-04 sec/m^3 for the TGB mixing case compared to 4.63E-04 sec/m^3 for the direct release to the atmosphere. This means that the fraction of the activity which enters the CR for the TGB mixing case will be:

$$F_{CR}\text{ (TGB mixing case)} = 0.0426\% \times (2.09E\text{-}04/4.63E\text{-}04) = 0.0192\%.$$

The CR dose for the case with mixing of the steam puff with the TGB air prior to release to the environment will be 0.45 times that of the case for the pure steam puff released directly to the atmosphere.

I.2 Calculation of the Source Term: The following assumption is applicable to the source term calculations:

Assumption B-1 The fission product inventory available for release is based on the reactor coolant DE I-131 concentration which is allowed by the Columbia Technical Specifications. To account for iodine spiking, the equilibrium level of DE I-131 is increased by a factor of 20.

Justification Per Section 15.6.4.5 of Reference 9, the only activity available for release from the MSLB is that present in the reactor coolant and steam lines prior to the break. This is consistent with the Technical Specifications as stated in Section 15.6.4.5 of Reference 9. Consistency with the current licensing basis is also maintained by the position that only the reactor coolant liquid contains the iodine. However, consistent with Reference 2, an increase by a factor of 20 will be included to take into account iodine spiking. This differs from the current analysis presented in Section 15.6.4.5 of Reference 9.

Based on Assumption B-1, the curie inventory of DE I-131 released in the MSLB is the product of the Technical Specification coolant activity concentration and the coolant liquid release from the break (105,000 lbm). Thus the coolant fission product inventory in Ci may be calculated as activity in μCi/gm x 105,000 lbm x 454 gm/lbm /1E6 μCi/Ci. Per Assumption B-1, the DE I-131 concentration (0.2 μCi/gm, as specified the Technical Specifications) is used in the calculation.

While it is true that a small amount of iodine may normally partition with the steam, the assumption that only the liquid coolant contains the iodine initially is more than compensated for by the iodine treatment in the dose calculation. In the dose calculation, it is assumed that all of the iodine contained in the liquid coolant (even the portion that does not flash) is added to the steam release puff.

As noted in Assumption B-1, there is a point of departure in this calculation relative to that reported in the FSAR; and that is the issue of iodine spiking. RG 1.183 (Reference 2) requires that iodine spiking be considered for analysis of MSLB dose at a value 20 times greater than the 0.2 μCi/gm value used for Section 15.6.4.5 of Reference 9. While the off-site dose limits from Reference 2 for MSLB without iodine spiking are a factor of 10 lower than the corresponding limits with spiking, the factor of 20 increase in activity outweighs the more favorable off-site dose limits by a factor of two. Moreover, for the CR dose, the limit for both cases is the same. Therefore, it is clear that the case with spiking is more limiting than the case without spiking.

and the spiking case is the only case explicitly analyzed. However, for completeness, both sets of results are presented (the spiking case doses are divided by a factor of 20 to obtain the non-spiking case results). The Technical Specification DE I-131 concentration multiplied by a factor of 20 (equivalent to 4 µCi/gm) is used for the spiking case.

I.3 Development of the STARDOSE model: The following assumptions are applicable to the STARDOSE model:

<u>Assumption B-2</u> The fission product release and other input parameters in STARDOSE are determined such that the amount of activity introduced into the CR (essentially instantaneously) is equal to the product of the time-integrated, normalized activity concentration (from the Reference 5 puff model, in sec/m^3), the activity released (Ci of DE I-131), and the volumetric flow of normal CR makeup (in m^3/sec). This activity is assumed to be instantaneously well-mixed within the CR volume.

<u>Justification</u> If the STARDOSE integrated activity introduced into the CR is equal to the actual time-integrated activity concentration (which results from the MSLB "puff" passing over the CR air intake) times the normal makeup rate, then the CR operators will be exposed to the correct source term. It is slightly conservative to introduce that amount of activity immediately, rather than over the two minute+ duration of the puff passage.

<u>Assumption B-3</u> For the CR MSLB dose calculation, no credit is taken for isolation and filtration of CR supply air.

<u>Justification</u> The normal makeup is assumed to continue for the duration of the accident. No credit is taken for a more rapid CR purge once the outside air concentration falls below that of the CR atmosphere.

The purpose of this subsection is to determine the STARDOSE model for fission product release which will provide a release equivalent to that from the diluted puff in terms of STARDOSE inputs. The results of this subsection are then used in the subsequent subsection to define the STARDOSE input file.

I.3.1 Activity Release: The STARDOSE fission product release is determined based on Assumption B-2 and is illustrated in Figure 1, page 4.000.

The DE I-131 total activity release is 4 µCi/gm x 105,000 lbm x 454 gm/lbm /1E6 µCi/Ci = 191 Ci. The LIBFILE1.TXT file for STARDOSE (Appendix A) has the inventory set at 4E-6 Ci, the coolant activity per gram obtained by dividing the total activity released (191 Ci) by the mass of liquid released (4.767E7 g). By design, the STARDOSE code expects inventories to be in Ci/MWt. To obtain the DE I-131 activity in the 105,000 lbm of coolant, the "power level" in the STARDOSE INPUT.DAT file must be set to 4.767E7, the number of grams of liquid coolant released.

The 0.0426% release (calculated above) to the CR is assumed to occur over six seconds (the steam release duration); therefore, the percent release per second is 0.0426% / 6 = 0.0071%/sec, and the percent release per hour is 0.0071%/sec x 3600 sec/hr = 25.6%/hr (or a fractional release rate per hour = 0.256). This results in a fractional release to the CR of 0.256 /hr x 6 sec /3600 sec/hr = 0.0426%, which is the fraction previously calculated in the I.1.2. The activity is then removed from the CR by being purged at the same rate of 1100 cfm as is assumed for the supply flowrate.

I.3.2 Control Room Model: The STARDOSE input file (INPUT.DAT) is presented in Appendix B. In this model, the CR control volume is set at 214,000 ft^3. The core control volume is as described above. Only two junctions are used: one from the core to the CR and one from the CR to the environment. No credit is taken for the CREF.

Dose conversion factors (which appear in the LIBFILE1.TXT file in Appendix A) are based on the default FGR 11 & 12 files in Reference 12 (consistent with the recommendations in References 1 and 2).

I.4 CR Dose Results and Conclusions: The STARDOSE-calculated doses include the effect of iodine spiking. To obtain "no spiking" doses, the "spiking included" iodine dose results were reduced by a factor of 20. It should be noted that neither filtered intake flow nor use of the remote intake(s) is credited in these results.

The results are summarized in Table 3 below.

Table 3. Control Room Dose From Direct Release to the Atmosphere

	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
Dose without iodine spiking (rem)	2.83E-05	8.95E-02	8.95E-02	5
Dose with iodine spiking (rem)	5.65E-04	1.79	1.79	5

This licensing-basis case can be confirmed by recognizing that (1) without filtration credit and (2) with a constant CR intake flow and exhaust rate, the normalized, time-integrated exposure within the control room would be the same as that at the CR air intake; i.e 8.19E-04 sec/m^3. If this value is multiplied by the 191 Ci DE I-131 released and by the breathing rate inside the CR (3.5E-04 m^3/sec), the result is the DE I-131 Ci inhaled (i.e., 5.5E-05 Ci). Since the CEDE dose conversion factor (approximately the same as the TEDE dose conversion factor since the whole body dose contribution is so small) for I-131 is 32893 rem/Ci inhaled, the corresponding dose would be 1.8 rem TEDE calculated as follows:

Inhaled Activity (Ci) = X/Q (sec/m^3) x Q (Ci) x Breathing Rate (m^3/sec)
= 8.19E-4 x 191 x 3.5E-4
= 5.5E-5 Ci

Dose (rem) = Inhaled Activity (Ci) x DCF (rem/Ci)
= 5.5E-5 x 32,893
= 1.8 rem

This confirms the STARDOSE calculation.

The above confirmation also provides a simple means of estimating the effect of reducing the intake flow/exhaust rate once activity has been brought into the CR at 1100 cfm. Since the activity decreases in the CR according to $e^{-\lambda t}$ (where λ is the fractional exhaust rate of the CR; i.e., 1100 cfm/214,000 ft^3 = 5.14E-03 /min) and since the integral to t = infinity for that expression is simply $1/\lambda$ (i.e., 194.5 minutes or 11,670 seconds), the normalized, time-integrated exposure within the CR is 11,670 seconds divided by the CR volume of 214,000 ft^3 or 6064 m^3. The result is 1.92 sec/m^3. Recalling that 0.000426 x 191 Ci DE I-131 were brought into the CR (i.e., 0.081 Ci), the time-integrated exposure is 1.92 sec/m^3 x 0.081 Ci = 0.156 Ci-sec/m^3. Mulitplying this value by the breathing rate of 3.5E-04 m^3/sec, one obtains the same 5.5E-05 Ci inhaled as in the previous paragraph. This means that the integrated exposure will increase inversely with the exhaust rate after the 0.081 Ci DE I-131 has been introduced. Therefore, if the intake/exhaust rate were 1100 cfm during the passage of the puff but were then reduced to 800 cfm, the CR dose would increase from 1.8 rem TEDE to 1100/800 x 1.8 rem TEDE = 2.5 rem TEDE.

The conclusion from these results is that the MSLB CR doses are below the 5.0 rem TEDE regulatory limit for control room operator exposure given in Reference 2 for BWR MSLB.

For the sensitivity case of pre-mixing the steam with 84% of the air in the TGB, the above doses have been reduced by a 0.45 multiplier; the results are presented in Table 4 below:

Table 4. Control Room Dose, Release Via TGB, Mixing w/ TGB Air

	Whole Body	CEDE	TEDE	Reg Limit (TEDE)
Dose without iodine spiking (rem)	1.27E-05	4.03E-02	4.03E-02	5
Dose with iodine spiking (rem)	2.54E-04	0.81	0.81	5

Even though these results in Table 4 have been compared to the regulatory limits, these are not considered licensing-basis results. They show only that the assumption of a steam-only puff (direct release to the environment rather than mixing with TGB air) is the limiting case (i.e., that any pre-dilution will reduce the dose even though the exposure duration may become longer).

II. Off-site Dose Calculation: Off-site doses include the Exclusion Area Boundary (EAB), and the Low Population Zone (LPZ). The following assumptions apply to the off-site dose calculations:

Assumption C-1 Same as Assumption B-1

Assumption C-2 Offsite dose assumes a direct unfiltered release to the environment; but because of the greater distances to the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary, the dispersed release is assumed to be a continuous plume, modeled with PAVAN, rather than a puff. As with the onsite (CR) dispersion, plume dilution due to buoyancy is not considered.

Justification This is based on the current licensing basis which assumes a direct unfiltered release to the environment with plume dispersion.

II.1 Dose Calculations: The following expression has been used to calculate the off-site doses:

Dose (rem) = Activity Release (Ci) x χ/Q (sec/m^3) x Breathing Rate (m^3/sec) x Dose Conversion Factor (rem/Ci)

The activity release (from the CR dose calculation above) is 191 Ci DE I-131. The χ/Q is 1.81E-4 sec/m^3 for the EAB and 4.95E-5 sec/m^3 for the LPZ. The breathing rate is 3.5E-4 m^3/sec. The I-131 CEDE DCF (from the CR dose calculation) is 32893 rem/Ci inhaled, and this is approximately the same for TEDE (because the whole body dose is negligible). Therefore, the doses are as follows:

Dose (EAB) = 191 Ci x 1.81E-4 sec/m^3 x 3.5E-4 m^3/sec x 32893 rem/Ci = 0.398 rem TEDE

Dose (LPZ) = 191 Ci x 4.95E-5 sec/m^3 x 3.5E-4 m^3/sec x 32893 rem/Ci = 0.109 rem TEDE

II.2 Offsite Dose Results and Conclusions: The dose results for MSLB with spiking considered are 0.398 rem TEDE for the EAB and 0.109 rem TEDE for the LPZ.

The conclusion from these results is that the MSLB offsite doses with spiking considered are well below the 25 rem TEDE regulatory limit from Reference 2 for BWR MSLB and spiking. In fact, these results are less than the 2.5 rem TEDE regulatory limit for BWR MSLB without spiking. Clearly, the Columbia MSLB is not of concern for offsite dose.

It is noted that the results from this calculation are consistent with the assumptions and inputs from Appendix D of RG 1.183 (Reference 2). This is evident from the following:

- The iodine concentration (with spiking) corresponds to 4.0 μCi/g DE I-131 in the reactor liquid coolant.
- The activity released from the fuel is assumed to mix homogeneously in the reactor coolant and is assumed to enter the steam phase instantaneously.
- Per Assumption A-1, the MSIV closure time (6 seconds) is assumed to be the maximum allowed time.
- Per Assumption A-1, total mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure. The steam release is equal to the sum of (1) the steam in the steam lines and connecting lines at the time of the break, (2) the steam which passes through the valves prior to closure, and (3) the steam which flashes from the liquid coolant which passes through the valves prior to closure.
- All radioactivity in the released liquid coolant is assumed to be released to the atmosphere instantaneously (Assumption A-1).
- All radioactivity in the released coolant is assumed to be released as a ground-level release (Assumption A-2).
- No credit is taken for plateout, holdup, or dilution within facility buildings for the primary license basis case (Assumption A-7).
- All iodine is assumed to partition with the gas (Assumption A-2) during the flashing of the liquid coolant which is very conservative relative to what would occur with the Reference 2, Appendix D, paragraph 4.4 specification. In reality, the bulk of the iodine would be expected to remain with the unflashed liquid coolant.

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Calculation of Pressure in the TGB

1- First consider the air in the TGB before the steam is added:

P1 V1 = n1 R T1

P1=1 atm = 14.7 psi
V1=5.7E+6 ft3 = 1.617E+11 cm3 = vol of TGB
R = 82.06 (atm.cm3)/(mol.K)
T1 = 25 C = 298 K = standered room temp.

n1 = 6.61E+6 moles of air exist in the TGB before the steam is added.

2- Consider the steam released:

mass of steam = 65,000 lbm = 2.95E+7 g = (2.95E+6 g)/(18 g/mole) = 1.64E+6 moles

3- Adding steam to the air in the TGB will result in a total number of moles of n2

n2 = Total no. of moles in the TGB = moles of air + moles of steam
n2 = 6.61E+6 + 1.64E=6 = 8.25E+6 moles

4- Average temperature of the air-steam mixture

Temp of steam = 342 F = 172 C = 445 K from MSLB calc, sec I.1.1.4
Temp of air in TGB = 25 C =298 K

T2 = [(6.61E+6 * 298) + (1.64E+6 * 445)]/8.25E+6
T2 = 327 K

5- Now applying the ideal gas law to the mixture

P2 v2 = n2 R T2

V2 = V1 = 5.7E+6 ft3 = 1.617E+11 cm3 = volume of TGB
R = 82.06 (atm.cm3)/(mol.K)
T2 = 327 K from step 4 above

P2 = 1.25 atm   the 0.25 atm = 3.67 psi (i.e; 0.25 * 14.7 psi/atm = 3.67 psi)

In fact, in the first two seconds of the puff injection into the TGB, there will be minimum mixing between the steam and the air in the TGB, this means that the steam plume will compress the air in the TGB and the air will compress the steam causing the initial pressure in the TGB to be even higher than the 3.67 psi calculated above using full mixing. The pressure of this isothermal compression of the two gases can be calculated as follows:

P1 V1 = P2 V2

P1 = 14.7 psi (initial pressure in the TGB before steam injection)
V1 = 5.7E+6 ft3 + 1.61E+6 ft3 = 7.3E+6 ft3 (Volume of the TGB + volume of puff before entering the TGB)
P2 = presuure in the TGB after the injection of the steam
V2 = volume of the TGB which will contain both gases (the two gases will be compressed in this volume)

P2 = P1 V1/V2
P2 = (14.7)(2.1E+6 + 5.7E+6)/(5.7E+6)

P2 = 20.1 psi
P2 – P1 = 20.1 – 14.7 = 5.4 psi is the pressure added to the TGB upon the injection of the steam before it mixes with air.

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Prepared by / Date: JD 6/17/04	Verified by/Date: BRA 6-18-04	Revision No. 0

LIBFILE1.TXT File for STARDOSE MSLB Run

```
n_isotopes 3 n_isotope_groups 11

I131Org Org_I   NONE  NONE  4e-6  9.96E-07  1.080E+06  6.734E-02  0  0  0.03  32893  0.13  0  0  0  0  0
I131Elem Elm_I  NONE  NONE  4e-6  9.96E-07  1.080E+06  6.734E-02  0  0  0.03  32893  0.13  0  0  0  0  0
I131Part Prt_I  NONE  NONE  4e-6  9.96E-07  1.080E+06  6.734E-02  0  0  0.03  32893  0.13  0  0  0  0  0
```

STARDOSE INPUT.DAT File

edit_time
0.0 24 720
end_edit_time

participating_isotopes
I131Org I131Elem I131Part
end_participating_isotopes

core

thermal_power	4.767e+007
elemental_iodine_frac	0.0485
organic_iodine_frac	0.0015
particulate_iodine_frac	0.95

release_frac
to_control_volume Control_Room

Time	N_Gas	I_Grp	CsGrp	TeGrp	BaGrp	NMtls	CeGrp	LaGrp	SrGrp
0.00167	0	0.256	0	0	0	0	0	0	0
720	0	0	0	0	0	0	0	0	0

end_to_control_volume
end_release_frac
end_core

control_volume

obj_type	OBJ_CR
name	Control_Room
air_volume	2.14e+005
water_volume	0
surface_area	0
has_recirc_filter	false

breathing_rate

Time(hr)	Value(cms)
720	0.00035

end_breathing_rate
occupancy_factor

Time(hr)	Value(frac)
24	1
96	0.6
720	0.4

end_occupancy_factor
end_control_volume

junction

junction_type	AIR_JUNCTION
downstream_location	AIR_SPACE
upstream	Core
downstream	Control_Room
has_filter	false

flow_rate

Time (hr)	Value (cfm)
720	1

REV. BAR

```
end_flow_rate
end_junction

junction
junction_type                AIR_JUNCTION
downstream_location          AIR_SPACE
upstream                     Control_Room
downstream                   environment
has_filter                   false
flow_rate
Time(hr)        Value(cfm)
720             1100
end_flow_rate
X_over_Q_4_ctrl_room
Time(hr)        Value(s/m*3)
720             0
end_X_over_Q_4_ctrl_room
X_over_Q_4_site_boundary
Time(hr)        Value(s/m*3)
720             0
end_X_over_Q_4_site_boundary
X_over_Q_4_low_population_zone
Time(hr)        Value(s/m*3)
720             0
end_X_over_Q_4_low_population_zone
end_junction

environment
breathing_rate_sb
Time (hr)       Value (cms)
8               0.00035
720             0.0
end_breathing_rate_sb
breathing_rate_lpz
Time (hr)       Value (cms)
8               0.00035
24              0.00018
720             0.00023
end_breathing_rate_lpz
end_environment
```

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		Calculation No. NE-02-04-06
Prepared by / Date: PT 6/17/04	Verified by/Date: BLH 6-18-04	Revision No. 0

STARDOSE RESULTS.OUT Excerpts

edit time 720.000000
Control_Room

	thyroid	wbody	skin	CEDE
Total dose:	5.87E+001	5.65E-004	4.66E-003	1.79E+000
Noble gas	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Org iodine	8.81E-002	8.47E-007	6.99E-006	2.68E-003
Elem iodine	2.85E+000	2.74E-005	2.26E-004	8.68E-002
Part iodine	5.58E+001	5.37E-004	4.43E-003	1.70E+000
Cesium	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Tellurium	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Barium	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Noble metal	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Lanthanides	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Cerium	0.00E+000	0.00E+000	0.00E+000	0.00E+000
Strontinum	0.00E+000	0.00E+000	0.00E+000	0.00E+000

	air_space	water_pool	surface	recirc	thyroid	wbody	skin	CEDE
I131Org	6.41E-097	0.00E+000	0.00E+000	0.00E+000	8.81E-002	8.47E-007	6.99E-006	2.68E-003
I131Elem	4.33E-096	0.00E+000	0.00E+000	0.00E+000	2.85E+000	2.74E-005	2.26E-004	8.68E-002
I131Part	3.89E-094	0.00E+000	0.00E+000	0.00E+000	5.58E+001	5.37E-004	4.43E-003	1.70E+000

STARDOSE 1.01 (c) 1996-2002 Polestar Applied Technology, Inc.
Thu Apr 01 09:38:21 2004

Purpose and Approach:

The purpose of this appendix is to provide an analysis of the Columbia MSLB using RADTRAD to check the main calculation. In this case, RADTRAD 3.02a has been used – it is expected that the results would be essentially the same for RADTRAD 3.03. Refer to the RADTRAD documentation (NUREG/CR-6604, main body Reference 12, and supplements) for a discussion of the relationship between RADTRAD 3.02a and 3.03.

In the main body of the calculation, the case with spiking was shown to release 191 Ci of dose-equivalent (DE) I-131. The puff X/Q was calculated to be 8.19E-4 sec/m^3. Finally, the transit time for the puff was calculated to be 127.6 seconds The effective volume dilution rate of the puff may be calculated from the inverse of the X/Q; i.e., 1/8.19E-4 m^3/sec = 1221 m^3/sec. The total volume dilution is the product of the effective volume dilution rate and transit time for the puff = 1221 m^3/sec x 127.6 sec = 1.558E5 m^3. The average puff concentration during its transit across the control room air intake may be calculated from this volume. With the known air intake rate and the known transit time, the RADTRAD analysis may be set up.

RADTRAD Analysis Calculation

The RADTRAD default input files for the iodine inventory and dose conversion factors (i.e., `c:\program files\u s nuclear regulatory commission\radtrad 3.02a\defaults\bwr_i131.nif`, and `c:\program files\u s nuclear regulatory commission\radtrad 3.02a\defaults\fgr11&12.inp`, respectively) were used. The default .nif file provides an iodine inventory per MWt of 0.2581E+05 Ci. Therefore, to release 191 Ci the .rtf file must be set up to release a fraction equal to 191/2.581E4 = 7.4E-3 as long as the power level is input as one MWt. This is the case, as seen in excerpts from the output file provided as Appendix D1 (these values are "boxed" so as to stand out in the input summary).

In RADTRAD, three compartments are used (see Appendix D1). The 191 Ci of DE I-131 are released to a "Plume" control volume with a volume of 5.5E6 ft^3 (same as 1.558E5 m^3) so that the correct average DE I-131 concentration is established. The control room air supply of 1100 cfm draws on this "Plume" control volume for 127.6 seconds (0.0354 hours) so that the correct amount of activity enters the control room. The control room volume is 214000 ft^3, as in the main calculation. The control room exhausts to the environment indefinitely at the same 1100 cfm rate. With a turnover rate of about 0.005 volumes per minute, the activity is cleared from the control room in about 3 x 1/0.005 minutes = 600 minutes = 10 hours. The problem is run for 24 hours.

The 191 Ci of DE-131 is also released from the "plume" to the environment at a rate of 1.44E5 %/day = 100% per minute. However, the release to the environment from the "Plume" control volume is kept at zero until 127.6 seconds (0.0354 hours) so that the control room dose is not affected. Because of the relatively long half-life of I-131, this delay has no effect on the dose calculation. The EAB dose X/Q is 1.81E-4 sec/m^3, the same value as in the main body of the calculation.

Results

The results are shown at the end of Appendix D-1. The doses are summarized as follows:

```
                  CR                    EAB
Time    Thyroid       TEDE      Thyroid       TEDE
(hr)     (rem)       (rem)       (rem)       (rem)
24.000 5.8236E+01 1.7736E+00 1.3068E+01 4.0017E-01
```

Conclusions

These doses agree well with the STARDOSE values of 1.79 rem TEDE for the control room and 0.398 rem TEDE for the EAB presented in the main body of the calculation. The RADTRAD analysis confirms the STARDOSE analysis.

ENERGY NORTHWEST People·Vision·Solutions	Appendix D1 – Excerpts from RADTRAD Output for MSLB	Page No. D1-1	Cont'd on page D1-2
		Calculation No. NE-02-04-06	
Prepared by / Date: [signature] 9/9/04	Verified by/Date: R-[signature] 9/14/04	Revision No. ~~0~~ 1 MAD 9/28/04	

REV. BAR

```
########################################################################
          RADTRAD Version 3.02a run on  6/02/2004  at  8:38:43
 ########################################################################

 ########################################################################
                      File information
 ########################################################################

 Plant file name           = C:\Program Files\U S Nuclear Regulatory
Commission\Radtrad 3.02a\Columbia_MSLB.psf
 Inventory file name       = c:\program files\u s nuclear regulatory
commission\radtrad 3.02a\defaults\bwr_i131.nif
 Scenario file name        = C:\Program Files\U S Nuclear Regulatory
Commission\Radtrad 3.02a\Columbia_MSLB.psf
 Release file name         = c:\program files\u s nuclear regulatory
commission\radtrad 3.02a\columbia_mslb.rft
 Dose conversion file name = c:\program files\u s nuclear regulatory
commission\radtrad 3.02a\defaults\fgr11&12.inp




     #####    ####   #####    #  #     #  #####   #    #   #####
     #    #  #       #        #  ##    #  #    #  #    #     #
     #    #  #       #        #  # #   #  #    #  #    #     #
     #####    ####   ####     #  #  #  #  #####   #    #     #
     #           #   #        #  #   # #  #       #    #     #
     #           #   #        #  #    ##  #       #    #     #
     #        ####   #        #  #     #  #        ####      #


 Radtrad 3.02 1/5/2000
 Columbia_MSLB
 Nuclide Inventory File:
 c:\program files\u s nuclear regulatory commission\radtrad
3.02a\defaults\bwr_i131.nif
 Plant Power Level:
  1.0000E+00
 Compartments:
   3
 Compartment 1:
 Plume
   3
  5.5000E+06
   0
   0
   0
   0
   0
 Compartment 2:
 CR
   1
  2.1400E+05
```

```
   0
   0
   0
   0
   0
 Compartment 3:
 E
   2
  0.0000E+00
   0
   0
   0
   0
   0
 Pathways:
   3
 Pathway 1:
 Plume to CR
   1
   2
   1
 Pathway 2:
 Plume to E
   1
   3
   4
 Pathway 3:
 CR to E
   2
   3
   1
 End of Plant Model File
 Scenario Description Name:

 Plant Model Filename:

 Source Term:
   1
   1   1.0000E+00
 c:\program files\u s nuclear regulatory commission\radtrad
3.02a\defaults\fgr11&12.inp
 c:\program files\u s nuclear regulatory commission\radtrad 3.02a\columbia_mslb.rft
   0.0000E+00
   0
   9.5000E-01   4.8500E-02   1.5000E-03   1.0000E+00
Overlying Pool:
   0
   0.0000E+00
   0
   0
   0
   0
 Compartments:
   3
```

REV. BAR

```
Compartment 1:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 2:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Compartment 3:
   0
   1
   0
   0
   0
   0
   0
   0
   0
 Pathways:
   3
 Pathway 1:
   0
   0
   1
   3
   0.0000E+00   1.0000E+00   1.1000E+03
   3.5400E-02   1.0000E+00   0.0000E+00
   2.4000E+01   1.0000E+00   0.0000E+00
   1
   3
   0.0000E+00   1.0000E+00   1.1000E+03
   3.5400E-02   1.0000E+00   0.0000E+00
   2.4000E+01   1.0000E+00   0.0000E+00
   1
   3
   0.0000E+00   1.0000E+00   1.1000E+03
   3.5400E-02   1.0000E+00   0.0000E+00
   2.4000E+01   1.0000E+00   0.0000E+00
   0
   0
   0
   0
```

```
   0
   0
   0
 Pathway 2:
   0
   0
   0
   0
   0
   0
   0
   0
   0
   0
   1
   3
   0.0000E+00   0.0000E+00
   3.5400E-02   1.4400E+05
   2.4000E+01   0.0000E+00
   0
 Pathway 3:
   0
   0
   1
   2
   0.0000E+00   1.0000E+00   1.1000E+03
   2.4000E+01   1.0000E+00   0.0000E+00
   1
   2
   0.0000E+00   1.0000E+00   1.1000E+03
   2.4000E+01   1.0000E+00   0.0000E+00
   1
   2
   0.0000E+00   1.0000E+00   1.1000E+03
   2.4000E+01   1.0000E+00   0.0000E+00
   0
   0
   0
   0
   0
   0
   0
 Dose Locations:
   2
Location 1:
 CR
   2
   0
   1
   2
   0.0000E+00   3.5000E-04
   2.4000E+01   0.0000E+00
   1
   2
```

```
   0.0000E+00   1.0000E+00
   2.4000E+01   0.0000E+00
Location 2:
 EAB
   3
   1
   3
   0.0000E+00   1.8100E-04
   2.0000E+00   0.0000E+00
   2.4000E+01   0.0000E+00
   1
   3
   0.0000E+00   3.5000E-04
   8.0000E+00   1.8000E-04
   2.4000E+01   0.0000E+00
   0
 Effective Volume Location:
   1
   2
   0.0000E+00   1.0000E+00
   2.4000E+01   0.0000E+00
 Simulation Parameters:
   1
   0.0000E+00   0.0000E+00
 Output Filename:
 C:\Program Files\U S Nuclear Regulatory Commission\Radtrad 3.o0
   1
   1
   1
   0
   0
 End of Scenario File

#######################################################################
          RADTRAD Version 3.02a run on   6/02/2004   at   8:38:43
 #######################################################################

 #######################################################################
                      Scenario Description
 #######################################################################

 Radioactive Decay is enabled
 RELEASE_NAME = Columbia_MSLB
 Release Fractions and Timings
                GAP         EARLY IN-VESSEL
             0.0003 hrs       0.0000 hrs
 NOBLES      0.0000E+00       0.0000E+00
 IODINE      7.4000E-03       0.0000E+00
 CESIUM      0.0000E+00       0.0000E+00
 TELLURIUM   0.0000E+00       0.0000E+00
 STRONTIUM   0.0000E+00       0.0000E+00
 BARIUM      0.0000E+00       0.0000E+00
 RUTHENIUM   0.0000E+00       0.0000E+00
 CERIUM      0.0000E+00       0.0000E+00
 LANTHANUM   0.0000E+00       0.0000E+00
```

REV. BAR

```
######################################################################
                    Cumulative Dose Summary
######################################################################
```

	CR		EAB	
Time (hr)	Thyroid (rem)	TEDE (rem)	Thyroid (rem)	TEDE (rem)
0.000	9.9475E-06	3.0295E-07	6.2315E-10	1.9083E-11
0.035	3.1832E-01	9.6943E-03	2.9963E-05	9.1757E-07
0.435	7.1078E+00	2.1647E-01	1.3066E+01	4.0012E-01
0.735	1.1672E+01	3.5547E-01	1.3066E+01	4.0013E-01
1.035	1.5828E+01	4.8204E-01	1.3067E+01	4.0014E-01
1.335	1.9613E+01	5.9731E-01	1.3067E+01	4.0015E-01
1.635	2.3060E+01	7.0228E-01	1.3067E+01	4.0016E-01
1.935	2.6198E+01	7.9787E-01	1.3068E+01	4.0017E-01
2.000	2.6837E+01	8.1730E-01	1.3068E+01	4.0017E-01
2.300	2.9638E+01	9.0261E-01	1.3068E+01	4.0017E-01
2.600	3.2189E+01	9.8030E-01	1.3068E+01	4.0017E-01
2.900	3.4512E+01	1.0510E+00	1.3068E+01	4.0017E-01
3.200	3.6627E+01	1.1155E+00	1.3068E+01	4.0017E-01
3.500	3.8553E+01	1.1741E+00	1.3068E+01	4.0017E-01
3.800	4.0308E+01	1.2276E+00	1.3068E+01	4.0017E-01
4.100	4.1905E+01	1.2762E+00	1.3068E+01	4.0017E-01
4.400	4.3360E+01	1.3205E+00	1.3068E+01	4.0017E-01
4.700	4.4685E+01	1.3609E+00	1.3068E+01	4.0017E-01
5.000	4.5891E+01	1.3976E+00	1.3068E+01	4.0017E-01
5.300	4.6990E+01	1.4311E+00	1.3068E+01	4.0017E-01
5.600	4.7990E+01	1.4615E+00	1.3068E+01	4.0017E-01
5.900	4.8901E+01	1.4893E+00	1.3068E+01	4.0017E-01
6.200	4.9731E+01	1.5145E+00	1.3068E+01	4.0017E-01
6.500	5.0486E+01	1.5376E+00	1.3068E+01	4.0017E-01
6.800	5.1174E+01	1.5585E+00	1.3068E+01	4.0017E-01
7.100	5.1801E+01	1.5776E+00	1.3068E+01	4.0017E-01
7.400	5.2372E+01	1.5950E+00	1.3068E+01	4.0017E-01
7.700	5.2891E+01	1.6108E+00	1.3068E+01	4.0017E-01
8.000	5.3364E+01	1.6252E+00	1.3068E+01	4.0017E-01
8.300	5.3795E+01	1.6383E+00	1.3068E+01	4.0017E-01
8.600	5.4187E+01	1.6503E+00	1.3068E+01	4.0017E-01
8.900	5.4545E+01	1.6612E+00	1.3068E+01	4.0017E-01
9.200	5.4870E+01	1.6711E+00	1.3068E+01	4.0017E-01
9.500	5.5166E+01	1.6801E+00	1.3068E+01	4.0017E-01
9.800	5.5436E+01	1.6883E+00	1.3068E+01	4.0017E-01
10.100	5.5682E+01	1.6958E+00	1.3068E+01	4.0017E-01
10.400	5.5906E+01	1.7026E+00	1.3068E+01	4.0017E-01
24.000	5.8236E+01	1.7736E+00	1.3068E+01	4.0017E-01

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1.0 DESCRIPTION

In accordance with 10 CFR 50.67, "Accident source term," a licensee may voluntarily revise the accident source term used in design basis radiological consequence analyses. Paragraph 50.67(b) requires that applications under this section contain an evaluation of the consequences of applicable design basis accidents (DBAs) previously analyzed in the plant Final Safety Analysis Report (FSAR). Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 1), provides guidance to licensees on performing evaluations and reanalyses as required to adopt an alternative source term (AST).

The AST is characterized by radionuclide composition and magnitude, chemical and physical form of the radionuclides, and the timing of the release of these radionuclides. An accident source term is a fundamental assumption upon which a portion of the plant design is based.

Energy Northwest has performed radiological consequence analyses of the four applicable boiling water reactor (BWR) DBAs identified in RG 1.183. These DBAs are a Loss of Coolant Accident (LOCA), a Fuel Handling Accident (FHA), a Control Rod Drop Accident (CRDA) and a Main Steam Line Break (MSLB). These analyses were performed using the guidance of RG 1.183 and Standard Review Plan (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms" (Reference 2). Comparison with the guidance contained in RG 1.183 is summarized in Attachment 2 of this license amendment request (LAR).

The supporting analyses consisted of the following steps:

- Determination of the AST based on plant-specific analysis of the fission product inventory,
- Application of the release fractions for the four BWR DBAs,
- Application of the deposition and removal mechanisms,
- Analysis of the atmospheric dispersion for the radiological propagation pathways, and
- Calculation of the offsite and control room (CR) personnel Total Effective Dose Equivalent (TEDE) doses.

In addition to revising the Columbia licensing basis to adopt the AST, licensing basis changes to the secondary containment drawdown and CR inleakage are proposed and justified to resolve existing non-conforming conditions associated with these two design functions.

2.0 PROPOSED CHANGE

The licensing and design basis changes included in this LAR are summarized in this section. The proposed Technical Specification (TS) changes are delineated below and a mark-up of the affected TS pages is provided in Attachment 3. A brief summary of the TS Bases changes is provided below and a mark-up of the affected pages is provided in Attachment 4 for information. Additionally, changes to the Columbia licensing and design basis are included in the LAR to resolve two previously identified nonconforming conditions. The first one is associated with secondary containment drawdown. This nonconforming condition has historically been referred to as the secondary containment drawdown Justification for Continued Operation (JCO). The second nonconforming condition is associated with CR inleakage. This nonconforming condition has historically been referred to as the CR inleakage unreviewed safety question (USQ). Brief summaries of these changes are provided below. Details of the analytical model used to resolve the secondary containment drawdown issue are provided in the Energy Northwest engineering calculation provided in Attachment 5. Additional details of these changes are provided in Section 3.0 "Background" and Section 4.0 "Technical Analysis." Numerous FSAR changes will be required based on the new analyses performed in support of this LAR. The updating of the FSAR to reflect these changes will be performed as part of the implementation of the LAR, and as such, the FSAR mark-ups are not provided with this submittal. The FSAR changes (not included) will be performed in accordance with 10 CFR 50.59.

<u>Technical Specification Changes</u>

Table of Contents

Deleted section 3.6.1.8, "Main Steam Isolation Valve Leakage Control (MSLC) System," and added section 3.9.10, "Decay Time." A discussion of the technical basis for these changes is provided below (see TS 3.6.1.8 and TS 3.9.10 change discussion).

TS 1.1, "Definitions"

Revised the definition for DOSE EQUIVALENT I-131 by replacing the word "thyroid" with "Total Effective Dose Equivalent (TEDE)" and replacing the references to dose conversion factors from TID-14844, RG 1.109, and ICRP-30, with a reference to Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

This change reflects the application of AST methodology.

TS 3.1.7, "Standby Liquid Control (SLC) System"

Added MODE 3 to the applicability statement and added the requirement to be in MODE 4 within 36 hours if a required action was not met.

This change is needed to support the use of the SLC system for buffering suppression pool pH as assumed in the LOCA analysis performed in support of this AST LAR.

Table 3.3.6.1-1, "Primary Containment Isolation Instrumentation"

Added MODE 3 to the applicable mode column for item k., "SLC System Initiation."

This change is needed for the reason stated above for TS 3.1.7.

Table 3.3.6.2-1, "Secondary Containment Isolation Instrumentation"

Deleted footnote (b) and corrected the spelling of "Function" in footnote (c).

Footnote (b) imposes operability requirements on the "Reactor Building Vent Exhaust Plenum Radiation – High" and the "Manual Initiation" functions during core alterations and fuel movements. Since secondary containment is not credited for the mitigation of the AST FHA, an operability requirement for these functions during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

The spelling correction is editorial.

TS 3.3.7.1, "Control Room Emergency Filtration (CREF) System Instrumentation"

Deleted Actions E and F.

Deleted "or radiation monitoring" and "as applicable" from Note 2 of the Surveillance Requirements (SR) section.

Actions E and F prescribe actions and completion times for an inoperable main CR ventilation radiation monitor. Entry into these two Actions is driven by item 4 of Table 3.3.7-1. Since item 4 of Table 3.3.7-1 is being deleted as discussed below, Actions E and F are no longer needed.

The radiation monitoring words deleted from the SR Note are associated with item 4 of Table 3.3.7-1 that is being deleted; therefore, these words are no longer needed.

Table 3.3.7.1-1, "Control Room Emergency Filtration (CREF) System Instrumentation"

Deleted footnote (b).

Deleted item 4, "Main Control Room Ventilation Radiation Monitor."

Footnote (b) imposes operability requirements on the "Reactor Building Vent Exhaust Plenum Radiation – High" (Table item 3) and the "Main Control Room Ventilation Radiation Monitor" (Table item 4) functions during core alterations and fuel movements. Since the CREF system is not credited for the mitigation of the AST FHA, an operability requirement for these functions during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

In addition to the above change, the remaining operability requirements (i.e., MODES 1, 2, 3 and during operations with a potential for draining the reactor vessel (OPDRV)) for the main CR ventilation radiation monitors are also obviated by the AST LOCA analysis associated with this LAR. While the operability of the CREF system is required for the AST LOCA analysis, the existing manual action for selecting the preferred remote intake based on the associated radiation levels as indicated by these monitors is no longer credited. Since this manual action is not credited, deletion of this TS function from this table is consistent with the criteria in 10 CFR 50.36.

TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)"

Deleted footnote 1 associated with SR 3.6.1.3.6.

Revised SR 3.6.1.3.10 to increase the allowable limit for secondary containment bypass leakage from 0.74 scfh to 0.04% primary containment volume/day.

Revised SR 3.6.1.3.11 to increase the allowable MSIV leakage limit from 11.5 scfh per valve to 16.0 scfh per valve when tested at greater than or equal to 25.0 psig.

The deletion of footnote 1 is editorial. This footnote was issued to address a special circumstance associated with a 2002 Notification of Enforcement Discretion and was limited to a specific time period that has expired.

The new allowable limits for bypass leakage and MSIV leakage are relaxations from the current requirements. The acceptability of these new limits is demonstrated in the supporting AST accident analyses. The resulting radiological consequences are within the applicable regulatory limits.

TS 3.6.1.8, "Main Steam Isolation Valve Leakage Control (MSLC) System"

Deleted entire TS.

This TS provided operability requirements for the MSLC system. This system is no longer credited for the mitigation of any DBA in the accident analyses performed in support of this AST LAR. Therefore, a TS requiring the operability of this system is no longer necessary and this deletion is consistent with the criteria of 10 CFR 50.36.

TS 3.6.4.1, "Secondary Containment"

Changes are proposed to the following three sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Action C related to fuel movement and core alterations. As a result of these deletions, Action C.3 became C.1. Additionally, the Limiting Condition for Operation (LCO) 3.0.3 note provided in Action C was deleted.

3) Revised SR 3.6.4.1.1 to change the minimum required containment vacuum from greater than or equal to 0.25 inch of vacuum water gauge to greater than 0.0 inch of vacuum water gauge. Deleted SR 3.6.4.1.4. Revised the existing SR 3.6.4.1.5 to change the maximum allowed standby gas treatment (SGT) subsystem flow rate from less than or equal to 2240 cubic feet per minute (cfm) to a secondary containment inleakage flow rate of less than or equal to 2430 cfm. Due to the deletion of SR 3.6.4.1.4, SR 3.6.4.1.5 is renumbered as SR 3.6.4.1.4.

This TS establishes the operability requirements for secondary containment. Since secondary containment is not credited for the mitigation of the AST FHA, the need to ensure the operability of this system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Action C.3 to C.1 is editorial. The LCO 3.0.3 note associated with Action C is no longer required.

The SR 3.6.4.1.1 supports the boundary condition (initial pre-accident pressure/vacuum) assumed for the air pressure in the secondary containment for the drawdown analysis. In NUREG-1434, (Reference 3) the specified vacuum value is bracketed and is dependent on the plant specific accident analysis. The new GOTHIC (Reference 4) model for secondary containment drawdown developed and presented in this LAR assumes an initial pressure that is based on a building pressure differential of 0.0 inches water gauge between the inside and the outside of the building at the bounding location. Therefore, the requirement to verify secondary containment vacuum is greater than or equal to 0.0 inches water gauge is appropriate for the new proposed licensing basis for secondary containment drawdown.

With the deletion of SR 3.6.4.1.4, existing SR 3.6.4.1.5 is renumbered as SR 3.6.4.1.4. The SR 3.6.4.1.4 currently requires secondary containment to be drawn down to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 120 seconds. This surveillance is no longer needed as secondary containment drawdown performance is adequately demonstrated by the proposed changes to existing SR 3.6.4.1.5 combined with proposed changes to SR 3.6.4.3.3. Taken together, these revised SRs provide a reasonable basis for demonstrating system operability and support AST LOCA analysis assumptions.

The maximum flow rate specified in SR 3.6.4.1.5 (new SR 3.6.4.1.4) has been revised to an inleakage flow rate of 2430 cfm. The revised value is equivalent to one secondary containment air volume exchange per day. This is consistent with the guideline in the SRP (Reference 5), Section 6.2.3. The revised value of 2430 cfm was calculated using the as-built secondary containment free volume.

New SR 3.6.4.1.4 verifies secondary containment integrity by ensuring that secondary containment inleakage does not prevent an acceptable drawdown. Revised SR 3.6.4.3.3 verifies that the SGT system reaches 4800 cfm within 2 minutes of an initiation signal. Performance of these surveillances provides assurance that secondary containment vacuum can be achieved and maintained.

TS 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Action D related to fuel movement and core alterations. As a result of these deletions, Action D.3 became D.1. Additionally, the LCO 3.0.3 note provided in Action D was deleted.

This TS establishes the operability requirements for SCIVs. Since secondary containment is not credited for the mitigation of the AST FHA, the need to ensure the operability of the SCIVs during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Action D.3 to D.1 is editorial. The LCO 3.0.3 note associated with Action D is no longer required.

TS 3.6.4.3, "Standby Gas Treatment (SGT) System"

Changes are proposed to the following three sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions C and E related to fuel movement and core alterations. As a result of these deletions, Action C.2.3 became C.2 and E.3 became E.1. Additionally, the LCO 3.0.3 notes provided in Actions C and E were deleted.

3) Revised SR 3.6.4.3.3 to add the phrase "and reaches greater than or equal to 4800 cfm within 2 minutes."

This TS establishes the operability requirements for SGT system. Since secondary containment is not credited for the mitigation of the AST FHA, the need to ensure the operability of the SGT system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Actions C.2.3 to C.2 and E.3 to E.1 are editorial. The LCO 3.0.3 notes associated with Actions C and E are no longer required.

The phrase "and reaches greater than or equal to 4800 cfm within 2 minutes" is an additional requirement that is proposed in this LAR. This new requirement supports the revisions to the SRs of TS Section 3.6.4.1. Establishing a flow rate acceptance criterion in SR 3.6.4.3.3 provides assurance that the SGT system performs at or above the level assumed in the secondary containment drawdown analysis. The 2-minute time period supports the bounding start time assumed in the drawdown analysis that considers a loss of offsite power, failure of the lead SGT fan to start and the subsequent autostart of the lag fan.

TS 3.7.3, "Control Room Emergency Filtration (CREF) System"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions D and F related to fuel movement and core alterations. As a result of these deletions, Actions D.2.3 became D.2 and F.3 became F.1. Additionally, the LCO 3.0.3 notes provided in Actions D and F were deleted.

This TS establishes the operability requirements for the CREF system. Since CREF is not credited for the mitigation of the AST FHA, the need to ensure the operability of the CREF system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Actions D.2.3 to D.2 and F.3 to F.1 are editorial. The LCO 3.0.3 notes associated with Actions D and F are no longer needed.

TS 3.7.4, "Control Room Air Conditioning (AC) System"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment, During CORE ALTERATIONS," from the applicability statement.

2) Deleted the portions of Actions C and E related to fuel movement and core alterations. As a result of these deletions, Actions C.2.3 became C.2 and E.3 became E.1. Additionally, the LCO 3.0.3 note provided in Actions C and E was deleted.

This TS establishes the operability requirements for the CR air conditioning system. The CR air conditioning system provides temperature control for the CR following isolation of the CR. Since CR isolation is not credited for the mitigation of the AST FHA, the operability of the CR air conditioning system during core alterations or fuel handling activities is no longer necessary. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Actions C.2.3 to C.2 and E.3 to E.1 are editorial. The LCO 3.0.3 note associated with Actions C and E is no longer needed.

TS 3.8.2, "AC Sources - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment" from the applicability statement.

2) Deleted the portions of Actions A and B related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1, A.2.4 became A.2.2, B.3 became B.1 and B.4 became B.2. Additionally, the LCO 3.0.3 note provided for the actions was deleted.

This TS establishes the operability requirements for AC sources during shutdown. Since no safety related systems are credited for the mitigation of the AST FHA, the requirement to ensure the operability of the supporting AC sources during core alterations and fuel handling activities is no longer needed. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

The existing requirements for the AC sources needed to support required equipment during MODES 4 and 5 are not relaxed by this change. This change aligns the scope of the applicability statement to be consistent with the above system specific TS changes relative to the revised requirements for fuel handling activities.

Changing Actions A.2.3 to A.2.1, A.2.4 to A.2.2, B.3 to B.1 and B.4 to B.2. are editorial. The LCO 3.0.3 note associated with these actions is no longer needed.

TS 3.8.5, "DC Sources - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment" from the applicability statement.

2) Deleted the portions of Action A related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1 and A.2.4 became A.2.2. Additionally, the LCO 3.0.3 note provided for this action was deleted.

This TS establishes the operability requirements for DC sources during shutdown. Since no safety related systems are credited for the mitigation of the AST FHA, the requirement to ensure the operability of the supporting DC sources during core alterations and fuel handling activities is no longer needed. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

The existing requirements for the DC sources needed to support required equipment during MODES 4 and 5 are not relaxed by this change. This change aligns the scope of the applicability statement to be consistent with the above system specific TS changes relative to the revised requirements for fuel handling activities.

Changing Actions A.2.3 to A.2.1 and A.2.4 to A.2.2 are editorial. The LCO 3.0.3 note associated with these actions is no longer needed.

TS 3.8.8, "Distribution Systems - Shutdown"

Changes are proposed to the following two sections of this TS:

1) Deleted "During movement of irradiated fuel assemblies in the secondary containment." from the applicability statement.

2) Deleted the portions of Action A related to core alterations and fuel movement. As a result of these deletions, Actions A.2.3 became A.2.1, A.2.4 became A.2.2 and A.2.5 became A.2.3. Additionally, the LCO 3.0.3 note provided for this action was deleted.

This TS establishes the operability requirements for the Division 1, 2 and 3 AC and DC electrical distribution system during shutdown. Since no safety related systems are credited for the mitigation of the AST FHA, the requirement to ensure the operability of the associated distribution systems during core alterations and fuel handling activities is no longer needed. This change, combined with the addition of TS 3.9.10, is consistent with the scope and intent of TSTF-51.

Changing Actions A.2.3 to A.2.1, A.2.4 to A.2.2, A.2.5 to A.2.3 are editorial. The LCO 3.0.3 note associated with this action is no longer needed.

TS 3.9.7, "Reactor Pressure Vessel (RPV) Water Level – New Fuel or Control Rods"

Changes are proposed to the following two sections of this TS:

Increased the required water level above the top of irradiated fuel assemblies seated within the RPV in the LCO from 22' to 23'.

Similarly, increased 22' to 23' in SR 3.9.7.1.

This change does not affect any accident analysis and does not affect the operation of the plant during refueling activities. This change is proposed to establish operational requirements consistent with assumptions of the AST FHA analysis.

A similar change to the existing water height requirement of 22' for TS 3.9.6, which is measured from the RPV flange, is not proposed. The physical dimensions of the RPV flange relative to maximum fuel pool water level preclude normal operation with water levels of 23'. However, the Bases for TS 3.9.6 has been changed as a result of the AST FHA analysis, and is included in Attachment 4.

TS 3.9.10, "Decay Time"

New TS 3.9.10 is proposed to ensure compliance with the decay time assumption used in the AST FHA analysis. This new TS requires a 24-hour decay time before in-vessel fuel movement can commence. A new SR is provided to verify compliance with the required decay time prior to the movement of irradiated fuel. A new TS Bases section is provided to discuss the applicable safety analysis and other supporting information.

Criterion 2 of 10 CFR 50.36(c)(2)(ii)(B) in part, specifies an operating restriction that is an initial condition of a design basis analysis as an item that should have a supporting LCO. A 24-hour decay time is assumed in the development of the source term used in the AST FHA analysis. This new TS is similar to the decay time TS proposed by the Tennessee Valley Authority in the recent Browns Ferry AST LAR (Reference 6). This change is consistent with the scope and intent of TSTF-51. The TSTF-51 specifies a decay time in the TS Bases. Columbia is proposing to specify this restriction as an LCO.

TS 5.5.7, "Ventilation Filter Test Program (VFTP)"

Revised the acceptable SGT system flow rates from a range of 4012 to 4902 cfm to a range 4320 to 5280 cfm in parts a, b, and d of this program description.

The new GOTHIC model for the secondary containment drawdown analysis credits a SGT system flow rate of 4800 cfm. The new 4800 cfm value for SGT system flow rate has been evaluated to ensure 99 percent filter efficiency credit in the design basis analyses. The change to the SGT system flow rate is an analytical change only. No changes to plant equipment or equipment setpoints are required. The proposed SGT system flow rate for filter test purposes is 4320 to 5280 cfm (i.e., 4800 ± 10%). This flow range complies with American National Standards Institute (ANSI) Standard N510-1989, "Testing of Nuclear Air Treatment Systems."

TS Bases Changes – Summary

The TS Bases were revised to incorporate results of the AST analyses. The reference sections were also updated. For example, numerous references to 10 CFR 100 were replaced with references to 10 CFR 50.67.

Secondary Containment Drawdown Licensing Basis Change

The original licensing basis for SGT system performance and the resulting secondary containment drawdown was based upon the ability of the SGT system to establish a 0.25 inch vacuum water gauge in the secondary containment within 120 seconds after a postulated LOCA. Based on a review of industry operating experience information in the late 1980s, Energy Northwest identified a condition outside the licensing basis under certain adverse conditions. Based on this condition, Energy Northwest developed an operability evaluation and submitted Revision 0 of a JCO to the NRC on September 29, 1989 (Reference 7). The JCO (currently Revision 6) assumes a 10-minute drawdown time.

Energy Northwest proposes to resolve this JCO by revising the design and licensing bases to a 20-minute drawdown time. This drawdown time is supported by a new calculation using a 3-node GOTHIC Version 7.1 model of the secondary containment and application of AST methodology. No equipment changes are required for either secondary containment or the SGT system.

Control Room Inleakage "USQ"

The original licensing basis for CR habitability assumed an unfiltered inleakage of 10.55 cfm. In response to an emerging generic industry concern, Energy Northwest performed a series of tracer gas tests in the fall of 2000 to assess the validity of the original inleakage assumption. Based on these tests, Energy Northwest determined this assumption could not be met and reported this condition in licensee event report (LER) 2000-006 (Reference 8). The impact on CR habitability was assessed and appropriate compensatory measures were established.

A second series of tracer gas tests were performed in the fall of 2003. The 2003 tests, performed by NUCON International, Inc., utilized the ASTM E741 methodology and current state-of-the-art testing technology. These tests provided more accurate results than the 2000 tests, but were still outside the original licensing basis.

The 2003 test results were used as the basis for the unfiltered inleakage assumptions in this LAR. Unfiltered CR inleakages of 75 cfm with both CREF trains in service and 50 cfm with one train in service were assumed. Both these values include 10 cfm for ingress and egress consistent with RG 1.197 (Reference 9).

3.0 BACKGROUND

Secondary Containment Drawdown JCO

This LAR provides the basis for resolving a long-standing nonconformance with the Columbia design and licensing bases regarding the establishment of secondary containment vacuum for mitigating DBAs. The original licensing basis for SGT system performance was to reestablish secondary containment to a 0.25 inch vacuum water gauge within 120 seconds of initiation after a DBA.

In 1988, the ability of the system to accomplish this design objective was brought into question. This was initially reported to the NRC staff in LER 88-023-00 (Reference 10). The NRC staff was notified of the interim resolution (JCO Revision 0) by a letter dated September 29, 1989 (Reference 7).

On January 3, 1990 (Reference 11) the NRC staff responded to the September 29, 1989 letter. That response acknowledged sufficient justification existed to allow continued operation. On February 16, 1990, Energy Northwest submitted a letter (Reference 12) to the staff that discussed a program plan for resolution of this issue.

On December 22, 1992, Energy Northwest submitted another letter (Reference 13) to the staff that discussed changes for the resolution of the secondary containment issue that was presented in the February 16, 1990 letter.

On October 15, 1996, Energy Northwest submitted the revised licensing basis and a request for amendment to secondary containment and SGT system TS (Reference 14). During the course of the NRC staff review of this amendment request, Energy Northwest responded to three Requests for Additional Information (RAIs) in letters dated December 4, 1997, April 12, 1999 and June 10, 1999 (References 15, 16 and 17).

On July 16, 1999, Energy Northwest withdrew the amendment request (Reference 18) due to a non-conservative error. On December 3, 2001 (Reference 19), Energy Northwest submitted a revised DBA analysis based on AST methodology to resolve the JCO. On November 20, 2002 (Reference 20) that submittal was withdrawn. The JCO is still in effect although it has been revised several times (currently Revision 6).

Energy Northwest proposes to resolve the nonconforming condition by revising the design and licensing basis. The resolution is supported by a new calculation using a 3-node GOTHIC model of the secondary containment and the application of the AST methodology for evaluating the associated radiological consequences. The proposed approach does not require hardware changes to either secondary containment or the SGT system.

<u>Control Room Boundary Inleakage</u>

This LAR also provides the basis for resolving a nonconformance with the design and licensing bases regarding unfiltered CR inleakage following a LOCA. The CR habitability systems are designed to maintain a suitable environment for plant operators during normal and abnormal operating conditions in accordance with General Design Criteria (GDC) 19 of 10 CFR 50, Appendix A. During a radiological accident, the CREF system provides protection for CR personnel by pressurizing the CR with filtered air drawn from two separate remote fresh air intakes.

In support of a previously submitted LAR (Reference 21) to adopt the AST, a test was performed in Fall 2000 to quantify unfiltered inleakage into the control room envelope (CRE). The results of the test showed CR inleakage was considerably higher than the 10.55 cfm assumed in the licensing basis. A follow-up operability assessment determined that this increased inleakage did not render the CREF system inoperable provided compensatory measures were implemented to administer potassium iodide (KI) to CR operators following a LOCA. This outside design basis condition was reported to the NRC in LER 2000-006-01 (Reference 8).

In response to Generic Letter 2003-01, Energy Northwest opted to re-perform the tracer gas testing. Results of this second series of tests are provided in Section 4.2 and are used to support this AST LAR.

4.0 TECHNICAL ANALYSIS

4.1 Secondary Containment Drawdown

Introduction

A new stand-alone analysis was performed to develop the revised design and licensing bases for secondary containment drawdown. An overview of this analysis is provided below. The approved calculation is provided in Attachment 5 of this submittal. A secondary containment drawdown time of 20 minutes is proposed as the new licensing basis. The analysis described below demonstrates the ability of the SGT system to support this licensing basis. The regulatory guidance provided in SRP Section 6.2.3 and RG 1.183 was used in the calculation. A 3-node GOTHIC Version 7.1 model was developed for this analysis.

Sensitivity studies were performed to develop a full understanding of the physics associated with the analysis. These sensitivity studies started with a single volume (i.e., 1-node) model. Additional modeling features such as internal heat sources, heat absorbing structures (thermal conductors), and wind pressure effects were added one at a time to ensure they were fully understood. These sensitivities provided additional insights and were used in the final 3-node model development.

Model Development and General Assumptions

The secondary containment was modeled as 3 nodes. This nodalization was selected to represent the major volumes within the reactor building.

The first of these nodes was the pump rooms located on the 422' 3" elevation. The pump rooms contain significant thermal heat loads and communicate with the suppression chamber of the primary containment via heat conduction.

The refueling floor was selected as the second node because it contains the spent fuel pool, a significant heat source, with a dedicated cooling system. Additionally, this volume included the upper reactor building siding, which is one of the potentially significant leakage paths.

The main building volume of the reactor building including the railroad bay up to the refueling floor was selected as the third node. The main building elevations are connected by a large open hatchway allowing relatively free exchange of air. This region includes a number of small rooms that contain heat sources and associated safety-related room coolers. Combining these rooms within this main building volume is conservative. Evenly distributing higher individual room heat effects throughout the entire main building volume reduces room temperature and the effectiveness of the associated room cooler to remove the heat. The air temperature reduction for the rooms artificially minimizes the temperature difference between the air and cooling water that services these coolers. Therefore, the heat removal provided by the coolers is under-predicted during the drawdown.

Only the Division 1 and 3 safety-related room coolers located in the reactor building were credited in the analysis as it was assumed the Division 2 diesel generator was unavailable. The overall heat transfer coefficients for these room coolers were assumed to remain constant for the 30-day mission time and were reduced to 60% to account for fouling and other variations.

Service water to the room coolers was assumed to be 78°F for the first two hours. This temperature bounds the 77°F verified every 24 hours by plant surveillance. The service water temperature was assumed to increase to a constant 87°F after the first two hours for the remainder of the time considered in the analysis. The value of 87°F bounds the average service water temperature for a 30-day LOCA analysis.

The initial air temperature of the volumes in the 3-node model was 75°F. The initial temperature of the volumes was based on an average reactor building temperature using plant operating experience during cold weather conditions. A low initial operating temperature increases drawdown time.

The room coolers are available to provide cooling 300 seconds after the start of the event. This time delay includes the sequence of events beginning with a loss of offsite power (LOOP), the starting and loading of the emergency diesel generators, and achieving full service water flow in the Emergency Core Cooling System (ECCS) room coolers.

The electrical heat loads assumed for the reactor building include high pressure core spray (HPCS), low pressure core spray (LPCS), residual heat removal (RHR) train A, and other loads supplied by Divisions 1 and 3. The emergency lighting heat loads were assumed to start at approximately 0 seconds and operate continuously. The heat load, associated with the normal operating equipment that is de-energized, was dissipated based on an exponential decay relationship.

The spent fuel pool decay heat load was assumed to be approximately 9.8E6 BTU/hr. This value is the typical maximum spent fuel pool heat load during startup expected after a refueling outage. Maintaining a constant value for the entire analysis period is conservative, as decay heat will decrease with time. Manual restoration of the fuel pool cooling system was assumed to occur at 12 hours following the initiation of a LOCA. A sensitivity analysis performed on the timing of this action showed the design function of achieving and maintaining a secondary containment vacuum is relatively insensitive to the timing. This is an existing manual action performed by procedures from the CR and is expected to occur well within the assumed 12 hours.

In accordance with SRP Section 6.2.3, no credit was taken for secondary containment outleakage.

The initial pressure inside secondary containment is established based on the 0.0 inch water gauge differential pressure between the inside and the outside of the reactor building at the limiting location. This assumption establishes the basis for the proposed change to TS SR 3.6.4.1.1 from 0.25 inch vacuum water gauge to 0.0 inch water gauge.

Bounding meteorological conditions were based on the extreme wind speed that is exceeded only 5% of the time. The 1996-1999 meteorological data used elsewhere in this submittal were the source for determining this wind speed. This wind speed is 17.2 mph at 33 feet. This is consistent with the guidance of RG 1.183.

The 5^{th} and 95^{th} percentile outside temperature values are 86°F on the high temperature side and 28°F on the low temperature side. Sensitivity studies were performed to determine which value to use for the bounding analysis.

For the purpose of this calculation, the SGT system fan was limited to a maximum of 4800 actual cubic feet per minute (acfm). The SGT system fan is assumed to start 120 seconds following the LOOP/LOCA.

The leakage flow split between the upper and lower elevations of the reactor building was based upon specific testing that was performed to determine the relative leakage at different locations in the building. Sensitivity analyses were performed to understand the effect of different flow split assumptions. As demonstrated in the sensitivity analysis, the 70/30 split (upper/lower) assumed in the analysis is conservative as compared to the 90/10 split suggested by the test data.

Sensitivity Analyses

Eight cases were evaluated as part of the model development to establish the necessary inputs for the design and licensing bases analysis. These cases address two wind directions, two outside temperature conditions, and two flow splits. A description of the cases is provided below.

Case 1: Warm air with easterly wind and 70/30 leakage flow split
Case 2: Warm air with south easterly wind and 70/30 leakage flow split
Case 3: Cold air with easterly wind and 70/30 leakage flow split

Case 4: Cold air with south easterly wind and 70/30 leakage flow split

Case 5: Warm air with easterly wind and 90/10 leakage flow split

Case 6: Cold air with easterly wind and 90/10 leakage flow split

Case 7: Warm air with south easterly wind and 90/10 leakage flow split

Case 8: Cold air with south easterly wind and 90/10 leakage flow split

These cases confirmed that the 70/30 split is conservative versus the 90/10 split for both warm (86°F) and cold (28°F) conditions. These cases also confirm that cold air case with easterly winds bound the other combinations of meteorological conditions. Therefore Case 3 was selected for development of the license basis model.

Using Case 3, four long-term cases were developed to evaluate the impact of delayed start of fuel pool cooling on maintaining a vacuum in secondary containment. The four cases are described below.

Case 9: Cold air easterly wind direction 70/30 leakage split fuel pool cooling start time 20 minutes

Case 10: Cold air easterly wind direction 70/30 leakage split fuel pool cooling start time 3 hours

Case 11: Cold air easterly wind direction 70/30 leakage split fuel pool cooling start time 12 hours

Case 12: Cold air easterly wind direction 70/30 leakage split fuel pool cooling start time 24 hours

All four cases demonstrated a vacuum could be maintained in secondary containment. The twelve hour case, Case 11, was chosen as a reasonable time to expect operator action to restore spent fuel pool cooling.

Conclusions and Results

This analysis demonstrated that the SGT system can restore and maintain secondary containment to at least 0.25 inches vacuum water gauge in less than 20 minutes. Based on this result a licensing basis drawdown time of 20 minutes was used in the LOCA analysis.

4.2 Control Room Boundary Inleakage

Introduction

In response to Generic Letter 2003-01, Energy Northwest performed tracer gas testing in Fall 2003. Test methodology, conditions, results, and the application of the results are discussed below.

Test Methodology

The tracer gas test was based on the constant injection method of ASTM E741-2000 (Reference 22). A constant flow of tracer gas is injected into the CRE until the resulting concentration in the envelope reaches a steady state (defined as exceeding a 95% approach to equilibrium). This occurs when the amount of tracer gas entering the CRE is the same as the amount leaving the CRE. By injecting the tracer gas in the air flow used for pressurization of the envelope, an estimate of the filtered and unfiltered airflow that provides this pressurization can be made by measuring the concentration of tracer gas in the airflow from the outside while at the same time measuring the steady state concentration in the CRE.

During performance of the inleakage tests, the CRE was administratively controlled to minimize casual ingress or egress. Measuring and test equipment were calibrated in accordance with the NUCON 10 CFR 50 Appendix B Quality Assurance (QA) program.

Description of the Columbia CRE

The CR is located on elevation 501' of the radwaste building. Included in the CRE are all essential control equipment of the plant plus a toilet, kitchenette, dining area, office area, and computer peripherals area. The CR is continuously occupied. The computer peripherals, kitchenette, and dining area are frequently occupied. The heating, ventilation and air conditioning (HVAC) equipment rooms (located on elevation 525' above the CR) are not in the CRE and are not serviced by the CR habitability systems. The CR HVAC equipment and associated ductwork necessary to preserve the unfiltered inleakage assumptions used in the dose analyses are included as part of the CRE.

Tests Conducted

Characterization Test: This test was performed while operating the CR HVAC system in the emergency pressurization mode with filter Train B in operation and the Division 1 (A) remote outside air intake open. The characterization test was performed to confirm that the CRE could be treated as a single zone. Approximately 30 minutes after the start of constant injection, gas samples were taken throughout the envelope. Analysis of the samples demonstrated that the spatial uniformity of tracer gas concentrations in the envelope differed by less than 10% from their average concentration. Based on these results, no additional fans for mixing were necessary and the CRE could be treated as a single zone.

Constant Injection Tests: The constant injection test with Train B in emergency pressurization mode operation was continued after the characterization test was performed. Constant injection tests on Train A were performed later, as well as both trains operating in parallel.

For the constant injection tests, the tracer gas concentration in the return airflow samples was monitored by taking samples approximately every 15 to 20 minutes until the concentration reached a steady state. All of the constant injection tests exceeded a 95% approach to equilibrium. Taking samples during this same interval also monitored tracer gas concentration in the airflow from the outside. The unfiltered inleakage airflows were calculated based on these tests and the results are shown in Table 4.2-1.

Uncertainties

Statistically based, random uncertainties were calculated with a 95% confidence level for the constant injection test. These results are shown in Table 4.2-1. As discussed in RG 1.197, for CREs that have low leakage (i.e., less than 100 cfm), the uncertainty may be an artifact of the calculations and not representative of CRE integrity.

Test Results

Table 4.2-1
CR Inleakage Test Results

Mode Tested	Unfiltered Inleakage
Train B Pressurization	8 ± 13 standard cubic feet per minute (scfm)
Train A Pressurization	-16 ± 26 scfm (effectively zero)
Train A Pressurization, 2nd Test	-26 ± 26 scfm (effectively zero)
Trains A + B Pressurization	27± 26 scfm

Conclusions

Based on the test results, 75 cfm of unfiltered air inleakage with both CREF trains in service and 50 cfm of unfiltered air inleakage with one CREF train in service were assumed in the AST LOCA analysis. These values include the allowance of 10 cfm for ingress and egress in accordance with the guidance in RG 1.197. Margin was added to the test result values to provide future operating margin. Given the small amount of measured leakage (i.e., less than 100 cfm), the margin provided is not intended to cover testing uncertainties. The exclusion of uncertainty in the license basis leakage values is acceptable per RG 1.197 for CRs with measured leakages of less than 100 cfm.

4.3 Atmospheric Dispersion Factors

Atmospheric dispersion factors (χ/Q) used in the LAR were calculated using plant specific meteorological data and the ARCON96 (Reference 23) and PAVAN (Reference 24) computer codes.

Meteorological Data

Certified meteorological data from the years 1996 through 1999 were used to calculate atmospheric dispersion factors to support this LAR. A CD-ROM of these data files is provided in Attachment 5 (see item 10 of Attachment 5). These four years of data were selected based on quality of the data, the quantity (i.e., recovery rate) of the data, and the representation of long term meteorological conditions and seasonal trends. The data set selected is consistent with RG 1.194 that states five years of hourly observations are considered representative of long-term trends at most sites and that one year including all four seasons is the minimum acceptable. The four-year data set used by Energy Northwest includes all four seasons for the four consecutive years in the data set and provides a representative long term trend. This conclusion is supported by a review performed by a certified meteorologist.

Energy Northwest upgraded much of the Columbia meteorological instrumentation in 2001. The reliability of the instrumentation during the period leading up to its replacement adversely affected the quantity and quality of the meteorological data collected in the years 2000 and 2001, thus these data were not included in the certified data set. The recovery rate of the 2002 data significantly improved and exceeded the 90% recovery rate standard described in Safety Guide 23 (Reference 25); however, some quality assurance issues were identified with the surveillances and calibration practices implemented with the installation of the new instrumentation. As a result of these quality issues, the 2002 data were not included in the certified data set.

The meteorological tower used for collecting the data is located less than 0.5 mile west of the plant site. Instrumentation is provided at the 33' level and the 245' level.

Calculation of Control Room χ/Qs

The ARCON96 computer code was used to calculate the CR χ/Qs, where χ is the concentration of a radionuclide at a receptor location in Ci/m^3-air normalized by the source emission rate Q in Ci/s. Five release points to the environment were modeled in the ARCON96 runs. These are:

1. The roofline source is an exhaust fan (short stack) on top of the reactor building at a height of 229' (70 m) above the ground through which routine releases take place. Following an accident, the exhausted air from the reactor building passes through the SGT filtration system before exiting through the roofline stack. This source is treated as a ground level point source in the χ/Q calculations.

2. The reactor building vehicle air lock doors (sometimes referred to as the King Kong, KK, doors) are located at the ground level on the eastside wall of the reactor building. The analysis assumes some leakage through these doors to the environment. The vehicle air lock doors are treated as a rectangular diffuse source that is 23' high x 20' wide.

3. The reactor building walls (RBWs) from the 606' level to the 670' level (top of reactor building) are made of metal sheets and are assumed to be a diffuse source capable of leaking radioactive materials to the atmosphere. This source is treated as a ground level release source.

4. The turbine building exhaust system (TBES) is a set of four circular exhaust fans (short stacks) located on top of the radwaste building roof. Air from the turbine building is exhausted to the atmosphere through these four fans. A rectangle was drawn around the four stacks. The closest point on the perimeter of this rectangle to the intake was then selected to calculate the distance between the source (one of the four exhaust fans) and the corresponding intake.

5. Two condensate storage tanks (CSTs), located north of the turbine building, have a potential to release radioactivity from liquid leakage originating from the suppression pool and bypassing the reactor building. (A short discussion of the χ/Q calculation for this source is provided at the end of this subsection. The CST calculation was performed separately from the above four release points.)

There are three intakes that can draw air into the CR. These are:

1. Local intake point: The local intake point is a louver located on the west side of the radwaste building wall at an elevation of 527' (26.5 m above the ground).

2. Remote intakes: There are two ground level remote intake points. Remote intake 1 is located northwest of the turbine building. Remote intake 2 is located southeast of the reactor building.

During normal operation all three are open. During post accident conditions when the CR is in the pressurization mode (i.e., post LOCA), the local intake is isolated and the two remote intakes remain open. Although isolated, some leakage flow through the local intake is conservatively assumed when calculating the χ/Qs for the CR.

Figures 4.3-1 and 4.3-2 provide a three-dimensional view and a plan view, respectively, of the relative locations of the sources and CR intakes.

Figure 4.3-1
Three Dimensional Source and Intake Locations
(not to scale – for illustrative purposes only)

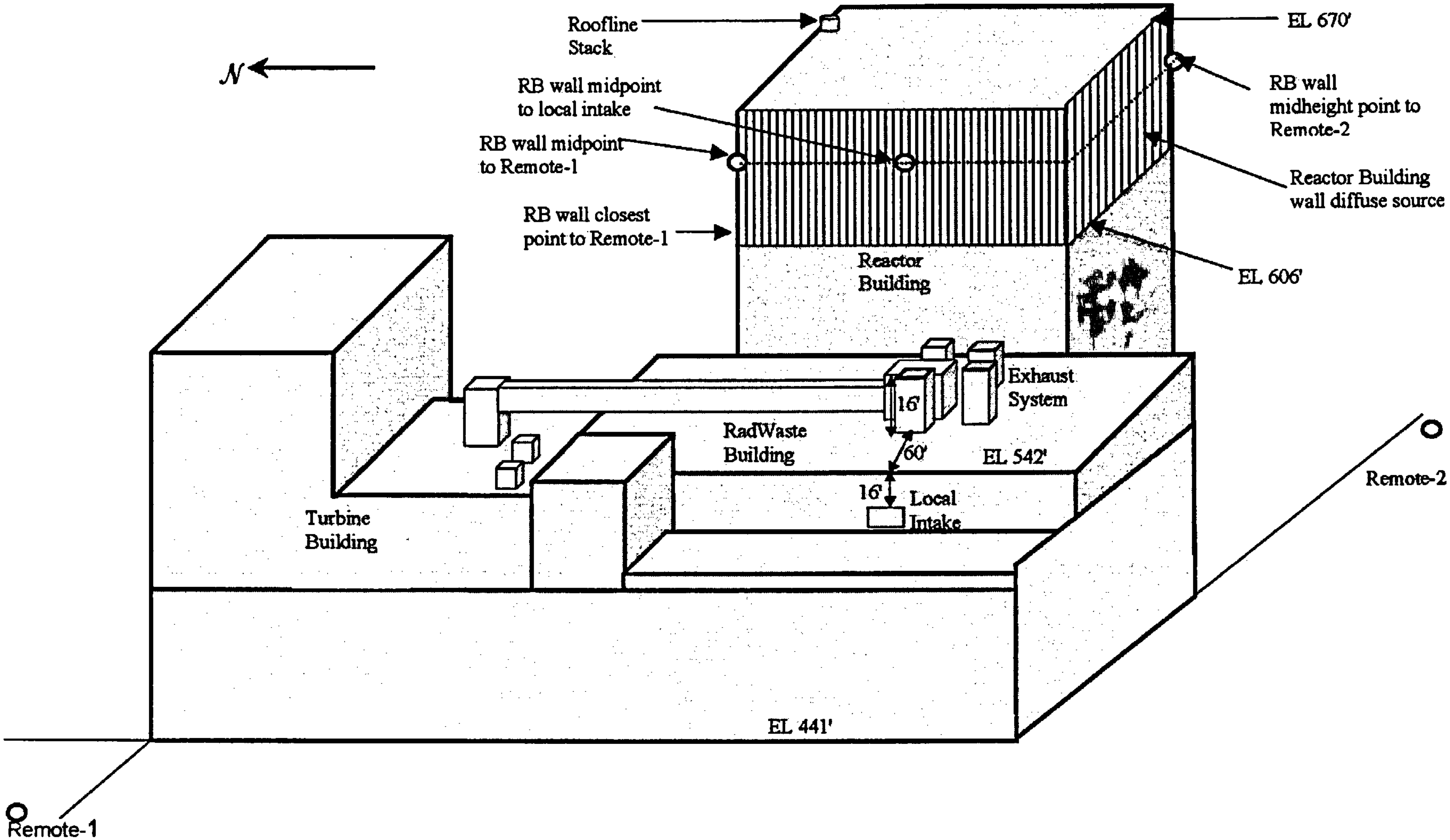
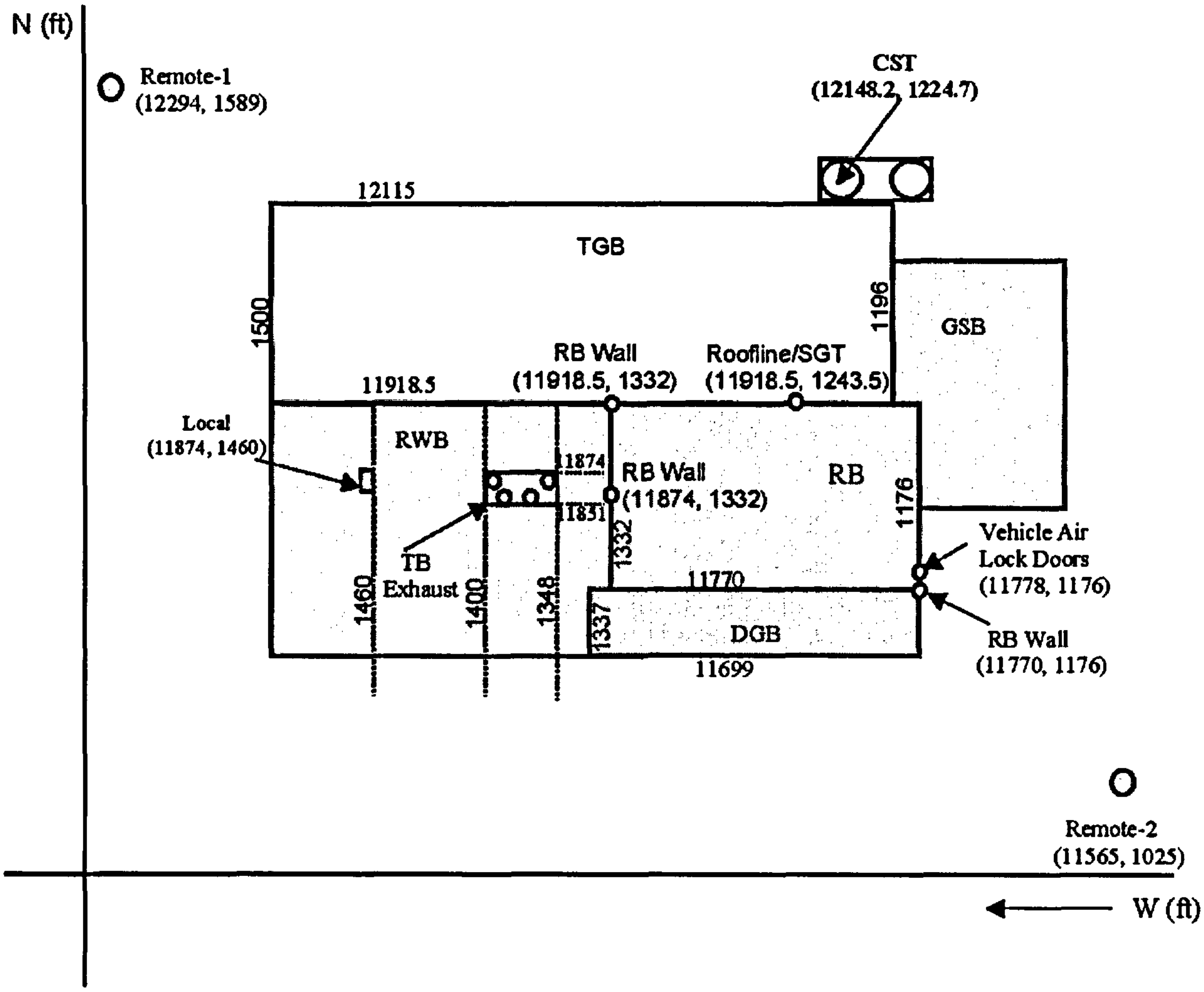


Figure 4.3-2
Plan View Source and Intake Locations
(not to scale – for illustrative purposes only)



Considering the first four sources and the three intakes, ARCON96 was run twelve times to address the various combinations.

The total filtered intake to the CR is a mixture from the three intakes. The RG 1.194 provides an equation for calculating an effective χ/Q for CRs with dual intakes. Using this guidance, an effective χ/Q was calculated for the three intakes.

1. Immediately following the design basis LOCA, the CR local intake is automatically secured and the CR pressurization process begins. Both trains of the CREF system receive a start signal and one or both start depending on whether a single failure of one train was postulated. The flow rate to the CR was measured under three test conditions: the usual surveillance testing, the system characterization testing, and the tracer gas testing. The difference in the intake flow rate results from different test conditions and flow measurement locations:

    - The surveillance testing uses a single train (either A or B) to draw air into the CR, while keeping both remote intakes open and the local intake closed. The flow rate was maintained between 900 and 1000 acfm.

    - The characterization testing showed that in dual train operation (both remotes open) the combined flow rate was 1544 acfm. In this same test, dual train operation with a single remote open, the combined flow rate was 1343 acfm. The local intake was secured during the tests.

    - The tracer gas testing used the alignment of two trains (A and B) to draw air into the CR, with a single remote intake open. The flow rate was greater than 1300 scfm. For a single train only, keeping one remote closed and the other open, the flow rate was greater than 800 scfm. The local intake was secured during the test.

    The effective χ/Qs were calculated using high and low bounding intake flow rates based on the above testing results. The worst-case effective χ/Qs were used in the LOCA analysis.

2. The local CR intake is assumed to leak air into the CR at a rate of 150 cfm of filtered leakage. This value is the leakage limit acceptance criterion for the intake dampers.

3. Since there are three CR intakes drawing air into the CR with different flow rates, equation 6b, Section 3.3.2.2 of RG 1.194, was used to calculate the effective χ/Q values. The use of this equation

was justified because no more than one intake can be within the 90-degree window from any release point. The 90-degree window is defined as a wedge centered on the line of sight between the source and the receptor with the vertex located on the release point, i.e., 45 degrees on either side of the line of sight. The equation has been slightly modified (as shown below) to account for the fact that there are three intakes instead of two:

$$\left(\frac{\chi}{Q}\right)_{eff} = \frac{\max\left[\left(\frac{\chi}{Q}\right)_L * F_L, \left(\frac{\chi}{Q}\right)_{R1} * F_{R1}, \left(\frac{\chi}{Q}\right)_{R2} * F_{R2}\right]}{F_L + F_{R1} + F_{R2}}$$

Where: L, R1, R2: denote the Local, Remote-1, and Remote-2 intakes, respectively, and

F: denotes the flow rate.

A summary of the ARCON96 input parameters for the first four of the five sources listed above is provided in Table 4.3-1.

Table 4.3-1
ARCON96 Input Parameters

Source →	Roofline (RL) Stack			King Kong (KK) Doors Secondary Containment (SC) Bypass			Reactor Building Walls (RBWs)			Turbine Building Exhaust (TBE)		
Receptor →	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2
Parameter	Sen-1 RL-L	Sen-2 RL-R1	Sen-3 RL-R2	Sen-4 KK-L	Sen-5 KK-R1	Sen-6 KK-R2	Sen-7 RBW-L	Sen-8 RBW-R1	Sen-9 RBW-R2	Sen-10 TBE-L	Sen-11 TBE-R1	Sen-12 TBE-R2
Meteorological Input												
Lower Met Tower Sensor Height (m)	10	10	10	10	10	10	10	10	10	10	10	10
Upper Met Tower Sensor Height (m)	75	75	75	75	75	75	75	75	75	75	75	75
Wind Speed Units	mph	mph	mph	mph	mph	mph	mph	mph	mph	mph	mph	mph
Receptor Input												
Distance to Receptor (m)	67.4	155.5	126.7	91.4	201.5	79.6	39	138.7	77.6	18.3	140.4	131.5
Intake Height Above Ground Level	26.5	0	0	26.5	0	0	26.5	0	0	26.5	0	0

Table 4.3-1
ARCON96 Input Parameters

Source →	Roofline (RL) Stack			King Kong (KK) Doors Secondary Containment (SC) Bypass			Reactor Building Walls (RBWs)			Turbine Building Exhaust (TBE)		
Receptor →	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2	Local	Rem-1	Rem-2
(m)												
Elevation Difference (m)	0	0	0	0	0	0	0	0	0	0	0	0
Direction to Source (deg)	78.39	137.38	328.28	108.68	141.33	324.67	90	145.61	323.6	90	155.77	311.5
Source Input												
Release Type	ground	ground	ground	ground	ground	ground	ground	ground	ground	ground	ground	ground
Release Height Above Ground Level (m)	70	70	70	3.5	3.5	3.5	60.0	60.0	60.0	36.3	36.3	36.3
Building X-sec area (m^2)	1787	2861	2861	1787	2861	2861	1787	2861	2861	1787	2861	2861
Vertical Velocity (m/s)	0	0	0	0	0	0	0	0	0	0	0	0
Stack Flow Rate (m^3/s)	2.1	2.1	2.1	0.0006	0.0006	0.0006	0.0006	0.0006	0.0006	55	55	55
Stack Radius (m)	0	0	0	0	0	0	0	0	0	0	0	0
Default Values												
Surface Roughness (m)	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
Wind Direction Window (deg.)	90	90	90	90	90	90	90	90	90	90	90	90
Minimum Wind Speed (m/s)	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5	0.5
Average Sector Width Constant	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3	4.3
Initial Diffusion Coefficients: Σ_y (m)	0	0	0	1	0.64	0.58	6.8	10.2	10.2	0.41	0.41	0.41
Initial Diffusion Coefficients: Σ_z (m)	0	0	0	1.16	1.16	1.16	3.25	3.25	3.25	0	0	0

The effective χ/Q results for filtered air intakes with one CREF train in operation at an assumed flow of 800 cfm are shown in Table 4.3-2. Values for 900 cfm were also calculated as shown in Attachment 5. The lower flow rate values resulted in higher doses.

The effective χ/Q results for filtered air intakes with dual CREF trains in operation at an assumed flow of 1300 cfm are shown in Table 4.3-3. Values for 1600 cfm were also calculated as shown in Attachment 5. The lower flow rate values resulted in higher doses.

The χ/Qs for the unfiltered air inleakage, taken directly from ARCON96, are shown in Table 4.3-2 and Table 4.3-2. These χ/Qs were calculated using local air intake as the receptor location. This receptor location was conservative as it resulted in higher χ/Qs than the two remote intakes. The χ/Qs for the unfiltered inleakage are the same in both tables because they are flow independent.

Table 4.3-2
Filtered CR Intake Flow of 800 cfm
(assuming single failure of one CREF train) and Unfiltered Inleakage χ/Q (s/m^3)

	Filtered				Unfiltered			
	Roofline Stack	KK doors SC Bypass	RBW SC Bypass	Turbine Building	Roofline Stack	KK doors SC Bypass	RBW SC Bypass	Turbine Building
0 - 2 hrs	1.43E-04	3.65E-04	1.99E-04	8.81E-04	6.95E-04	5.34E-04	8.69E-04	4.70E-03
2 - 8 hrs	1.05E-04	2.89E-04	1.44E-04	3.75E-04	3.36E-04	1.97E-04	4.40E-04	2.00E-03
8 - 24 hrs	4.14E-05	1.18E-04	5.73E-05	1.93E-04	1.28E-04	8.41E-05	1.75E-04	1.03E-03
1 - 4 days	3.52E-05	9.83E-05	5.00E-05	1.50E-04	9.72E-05	7.26E-05	1.38E-04	8.01E-04
4 - 30 days	3.03E-05	8.61E-05	4.18E-05	1.44E-04	7.69E-05	7.00E-05	1.10E-04	7.69E-04

Table 4.3-3
Filtered CR Intake Flow of 1300 cfm
(assuming Both Trains Remain on For 30 Days) and Unfiltered Inleakage χ/Q (s/m^3)

	Filtered				Unfiltered			
	Roofline Stack	KK doors SC Bypass	RBW SC Bypass	Turbine Building	Roofline Stack	KK doors SC Bypass	RBW SC Bypass	Turbine Building
0 - 2 hrs	1.56E-04	3.98E-04	2.17E-04	5.42E-04	6.95E-04	5.34E-04	8.69E-04	4.70E-03
2 - 8 hrs	1.15E-04	3.15E-04	1.57E-04	2.31E-04	3.36E-04	1.97E-04	4.40E-04	2.00E-03
8 - 24 hrs	4.51E-05	1.28E-04	6.24E-05	1.19E-04	1.28E-04	8.41E-05	1.75E-04	1.03E-03
1 - 4 days	3.83E-05	1.07E-04	5.44E-05	9.24E-05	9.72E-05	7.26E-05	1.38E-04	8.01E-04
4 - 30 days	3.30E-05	9.38E-05	4.56E-05	8.87E-05	7.69E-05	7.00E-05	1.10E-04	7.69E-04

A χ/Q for the CST source is needed for the calculation of the radiation dose due to the secondary containment liquid leakage bypass. An additional ARCON96 run was performed to determine the χ/Q for the CST source. The CST is a set of two tanks located north of the turbine building. The CR remote intake 1 is the closest of the three intakes to the CST source. For conservatism, the χ/Q calculation assumed the receptor intake was the remote intake 1. The results of the ARCON96 run are in Table 4.3-4.

Table 4.3-4 χ/Q Values from the CST to Remote-1 Intake	
Time Period	**χ/Q (s/m^3)**
0 - 2 hrs	4.18E-04
2 - 8 hrs	1.59E-04
8 - 24 hrs	6.31E-05
1 - 4 days	5.78E-05
4 - 30 days	5.57E-05

<u>Calculation of Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) χ/Qs</u>

The PAVAN computer code was used to calculate the χ/Q values for the EAB and LPZ. This methodology is consistent with RG 1.145 (Reference 26).

The following data were used as input to PAVAN.

1. Since the roofline stack is not two and one-half times higher than adjacent buildings, the ground level release mode was used.
2. Distance to the EAB is 1950 m.
3. Distance to the LPZ is 4827 m.
4. Reactor building height is 69.8 m.
5. Reactor building cross-sectional area is 2861 m^2 calculated using the smallest width of the wall.

6. The four hourly joint frequency data (JFD) files for the years 1996-1999 were added to generate a single hourly JFD file representing that period of time. Eleven wind-speed categories were used in those JFDs.

7. The calm wind category was distributed separately from the other eleven wind speed categories.

8. The option to use both desert sigma and Pasquill - Gifford sigma was activated in PAVAN, then the highest χ/Q was selected.

9. The default terrain adjustment factor was used.

PAVAN uses three procedures to calculate χ/Q for the EAB and LPZ.

1. The 0.5-percent procedure

2. The SRP 2.3.4 procedure, and

3. The 5-percent site limit procedure.

Consistent with RG 1.145 only two of the three PAVAN procedures (1 and 3) were used. The results were compared and the χ/Q values from the 0.5-percent procedure were slightly higher than those from the 5-percent site-limit procedure; therefore, the 0.5-percent χ/Q values were selected. Table 4.3-5 summarizes the results of χ/Q values calculated with the PAVAN computer code.

Table 4.3-5
PAVAN Analysis Results

Time Period	EAB χ/Q (s/m^3)	LPZ χ/Q (s/m^3)
0 - 2 hrs	1.81E-4	-
0 - 8 hrs	-	4.95 E-5
8 - 24 hrs	-	3.69 E-5
1 - 4 days	-	1.95 E-5
4 - 30 days	-	7.81 E-6

4.4 Loss of Coolant Accident

4.4.1 Introduction and Background

Columbia is a BWR/5 with a Mark II containment. The rated power is 3486 MWt. This value is increased by 2% to 3556 MWt in the analysis described below to account for power measurement uncertainties. The core inventory used to develop the source term for the LOCA analysis is based on an adjusted plant-specific pre-1995 ORIGEN 2 run. The Columbia Mark II containment consists of two compartments. The two compartments are connected by a vent system that allows steam released from the reactor vessel (located in the drywell) to flow into the suppression pool. Drywell sprays are credited for reducing primary containment pressure and scrubbing the drywell atmosphere. Manual initiation of drywell sprays is assumed to occur 15 minutes after the LOCA. Primary containment leakage is limited by TS to 0.5% volume per day. Because of post-accident containment depressurization, this leakage rate will decrease with time. A factor of two reduction in the leak rate after 24 hours is assumed in this analysis. Prior to the completion of the secondary containment drawdown, the containment leakage is assumed to go directly to the environment. After the 20-minute drawdown period, filtration of the leakage is credited; however, no credit is taken for holdup in secondary containment.

Two sources of containment leakage that bypass the secondary containment are the MSIV leakage and the miscellaneous bypass leakage. The proposed TS allowable MSIV leakage limit of 16.0 scfh per valve (adjusted for peak accident containment pressure) was assumed. For secondary containment bypass leakage, the proposed TS allowable limit of 0.04% primary containment volume per day was assumed. These leakage rates are reduced by a factor of two after 24 hours because of post-accident containment depressurization.

Natural deposition of radioactive particulates is credited for three of the four main steam lines. Since a single failure of an outboard MSIV in one steam line is assumed, natural deposition is not credited in this line.

Maintaining the suppression pool pH above 7.0 serves to improve its iodine retention capability and reduces the amount of radioactive iodine available for release in the design basis LOCA. Buffering of the suppression pool by the SLC system is credited to maintain the pH of the suppression pool above 7.0. The initiation of the SLC system is a manual action.

The radiological dose to the CR operators during the postulated design basis LOCA is mitigated by the integrity of the CRE and operation of the CREF system. The doses calculated in this AST evaluation are based on the limiting combinations of unfiltered leakages and filtered intake flows coupled with conservatively selected χ/Qs.

The STARDOSE computer code is used to calculate the dose to the CR operator as well as the doses at the EAB and LPZ.

The bounding radiological analysis for the LOCA event detailed in this section reflects an inadequate core cooling accident that degrades to core damage. Unlike the current licensing basis, this event is not prescribed as a mechanistic double-ended guillotine break of the recirculation system pump suction piping.

The key parameters used in the design basis AST LOCA analysis are listed in Table 4.4-1.

Table 4.4-1
Key Parameters for AST LOCA Analysis

Columbia Design Input Parameter	**Parameter Value**
Core power	3556 MWt
Secondary containment drawdown time	20 minutes
Drywell spray initiation time	15 minutes
Volumetric flow rate, drywell to environment (Non-MSIV)	0.54% drywell volume per day (secondary containment bypass before drawdown) 0.04% drywell volume per day (secondary containment bypass after drawdown)
Volumetric flow rate, wetwell to environment	0.54% wetwell volume per day (secondary containment bypass before drawdown) 0.04% wetwell volume per day (secondary containment bypass after drawdown)
Volumetric flow rate, drywell to secondary containment	0% drywell volume per day (before drawdown) 0.5% drywell volume per day (after drawdown)
Volumetric flow rate, wetwell to secondary containment	0% wetwell volume per day (before drawdown) 0.5% wetwell volume per day (after drawdown)
Volumetric flow rate, secondary containment to environment through	5000 cfm (before drawdown) 5000 cfm (after drawdown)

Table 4.4-1
Key Parameters for AST LOCA Analysis

SGT	
Volumetric flow rate, secondary containment to environment bypassing SGT filters	50 cfm (after drawdown)
Volumetric flow rate, ESF leakage into secondary containment	1 gpm (analyzed as 2 gpm)
Volumetric flow rate, drywell to environment via the main steam lines	Based on 16 scfh at test pressure of ≥ 25 psig per valve
Filter efficiencies for SGT	0% for all species before drawdown 99% for all species except noble gases after drawdown 0% for noble gases
Filter efficiencies for CREF	99% for particulates 95% for elemental and organic iodines 0% for noble gases
Volume of CR	214,000 ft^3
CR occupancy factor	0 - 24 hrs: 1 1 - 4 days: 0.6 4 - 30 days: 0.4
Breathing rate (CR)	0 – 30 days: 3.5E-4 m^3/sec
CREF filtered intake flow	Single CREF: 800 cfm (minimum) Both CREF: 1300 cfm (minimum)
CR unfiltered flow	Single CREF: 50 cfm Both CREF: 75 cfm
Breathing rate (both EAB and LPZ)	0 - 8 hrs: 3.5E-4 m^3/sec 8 - 24 hrs: 1.8E-4 m^3/sec 1 - 30 days: 2.3E-4 m^3/sec
Effective χ/Qs for CR	See Tables 4.4-2 and 4.4-3
χ/Q, CST	See Table 4.3-4
χ/Q, EAB and LPZ	See Table 4.3-5
Dose conversion factors	Based on FGR 11 and FGR 12 (defaults for RADTRAD)

Table 4.4-2
Effective χ/Qs (sec/m^3) for Control Room with 800 cfm intake flow (single-train CREF, min flow)
(based on values from Table 4.3-2)

	Filtered			Unfiltered		
Time Frame	Turbine Building	SC Bypass*	Roofline Stack	Turbine Building	SC Bypass*	Roofline Stack
0 - 2 hrs	8.81E-04	2.82E-04	1.43E-04	4.70E-03	7.02E-04	6.95E-04
2 - 8 hrs	3.75E-04	2.17E-04	1.05E-04	2.00E-03	3.19E-04	3.36E-04
8 - 24 hrs	1.93E-04	8.77E-05	4.14E-05	1.03E-03	1.30E-04	1.28E-04
1 - 4 days	1.50E-04	7.42E-05	3.52E-05	8.01E-04	1.05E-04	9.72E-05
4 - 30 days	1.44E-04	6.40E-05	3.03E-05	7.69E-04	9.00E-05	7.69E-05

* Average of "KK doors SC bypass" and "RBW SC bypass"

Table 4.4-3
Effective χ/Qs (sec/m^3) for Control Room with 1300 cfm intake flow (two-train CREF, min flow)
(based on values from Table 4.3-3)

	Filtered			Unfiltered		
Time Frame	Turbine Building	SC Bypass*	Roofline Stack	Turbine Building	SC Bypass*	Roofline Stack
0 - 2 hrs	5.42E-04	3.08E-04	1.56E-04	4.70E-03	7.02E-04	6.95E-04
2 - 8 hrs	2.31E-04	2.36E-04	1.15E-04	2.00E-03	3.19E-04	3.36E-04
8 - 24 hrs	1.19E-04	9.52E-05	4.51E-05	1.03E-03	1.30E-04	1.28E-04
1 - 4 days	9.24E-05	8.07E-05	3.83E-05	8.01E-04	1.05E-04	9.72E-05
4 - 30 days	8.87E-05	6.97E-05	3.30E-05	7.69E-04	9.00E-05	7.69E-05

* Average of "KK doors SC bypass" and "RBW SC bypass"

4.4.2 Source Term

The source term used for the design basis LOCA analysis is defined by the quantity, type, and timing of the release of radioactivity from a damaged reactor core to the containment. The core inventory is provided in Table 4.4-4 and the release rates are shown in Table 4.4-5. These inventories are based on an adjusted plant-specific pre-1995 ORIGEN 2 run. The three adjustments were:

- A scale factor to bound the power level to 3556 MWt,
- A correction to increase selected krypton values (based on comparisons to other core inventory tables), and
- An increase in the activity of longer lived isotopes.

These adjustments resulted in a conservative source term (in terms of activity available). The assumed core power of 3556 MWt is the licensed power increased by 2% to account for power measurement uncertainties in accordance with SRP Section 15.6.5 (Reference 27).

Table 4.4-4
Core Inventory at Time Zero

Nuclide	Ci/MWt	Nuclide	Ci/MWt	Nuclide	Ci/MWt
Kr83m	3.57E+03	I134Part	6.03E+04	Y93	3.56E+04
Kr85m	7.35E+03	I135Part	5.03E+04	Zr95	4.27E+04
Kr85	4.11E+02	Rb86	4.47E+01	Zr97	4.33E+04
Kr87	1.34E+04	Cs134	6.27E+03	Nb95	4.27E+04
Kr88	1.90E+04	Cs136	1.39E+03	La140	4.71E+04
Kr89	2.20E+04	Cs137	5.05E+03	La141	4.36E+04
Xe131m	2.79E+02	Sb127	3.31E+03	La142	4.17E+04
Xe133m	1.66E+03	Sb129	9.48E+03	Pr143	3.78E+04
Xe133	5.43E+04	Te127m	4.66E+02	Nd147	1.71E+04
Xe135m	1.11E+04	Te127	3.31E+03	Am241	7.67E+00
Xe135	1.31E+04	Te129m	1.39E+03	Cm242	1.74E+03
Xe137	4.65E+04	Te129	8.90E+03	Cm244	1.41E+02
Xe138	3.59E+04	Te131m	4.20E+03	Ce141	4.43E+04
I131Org	2.79E+04	Te132	3.99E+04	Ce143	4.01E+04
I132Org	3.94E+04	Ba137m	3.01E+03	Ce144	3.25E+04
I133Org	5.44E+04	Ba139	4.72E+04	Np239	7.01E+05
I134Org	6.03E+04	Ba140	4.58E+04	Pu238	9.56E+01
I135Org	5.03E+04	Mo99	4.90E+04	Pu239	1.89E+01
I131Elem	2.79E+04	Tc99m	4.34E+04	Pu240	3.11E+01
I132Elem	3.94E+04	Ru103	4.70E+04	Pu241	8.85E+03
I133Elem	5.44E+04	Ru105	3.46E+04	Sr89	2.02E+04
I134Elem	6.03E+04	Ru106	2.04E+04	Sr90	3.34E+03
I135Elem	5.03E+04	Rh105	3.27E+04	Sr91	2.59E+04
I131Part	2.79E+04	Y90	2.04E+03	Sr92	3.01E+04
I132Part	3.94E+04	Y91	2.73E+04		
I133Part	5.44E+04	Y92	2.90E+04		

Table 4.4-5
Release Rates For The Core Inventory*

Release Phase	Fraction of Core Inventory Released			
			Per Hour	Total
0 – 0.033 hours	No Release			
Gap release 0.033 – 0.533 hours	Gases	Xe, Kr Elemental I Organic I	1.0E-1/hr 4.9E-3/hr 1.5E-4/hr	5.0E-2 2.4E-3 7.5E-5
	Aerosols	I, Br Cs, Rb	9.5E-2/hr 1.0E-1/hr	4.8E-2 5.0E-2
Fuel release 0.533 –2.033 hours	Gases	Xe, Kr Elemental I Organic I	6.3E-1/hr 8.1E-3/hr 2.5E-4/hr	9.5E-1 1.2E-2 3.8E-4
	Aerosols	I, Br Cs, Rb Te Group Ba, Sr Noble Metals La Group Ce Group	1.6E-1/hr 1.3E-1/hr 3.3E-2/hr 1.3E-2/hr 1.7E-3/hr 1.3E-4/hr 3.3E-4/hr	2.4E-1 2.0E-1 5.0E-2 2.0E-2 2.5E-3 2.0E-4 5.0E-4

* Consistent with RG 1.183, two core inventory release phases were modeled following a 120 second (0.033 hours) delay.

4.4.3 Mitigation

The radiological consequences of the LOCA are actively mitigated by several safety-related systems. The CREF system is credited for the mitigation of the dose to the CR operator. The isolation of the CR and the initiation of the CREF system are automatic in response to an accident (FAZ) signal.

- F High Drywell Pressure
- A Low-Low Reactor Water Level
- Z High Radiation Reactor Building Exhaust

Both CR remote intakes are normally open. For the licensing basis case, only one train of CREF is credited. From the CR inleakage USQ discussion above, 50 cfm of unfiltered inleakage is assumed for a single CREF train scenario. Filtered intake flow of 800 cfm for CR pressurization is assumed with CREF filter efficiencies of 95% for the gaseous iodine species and 99% for the particulates. No manual actions are credited in the analysis relative to the CREF system.

The SGT system is credited for the mitigation of the radiological releases. Credit for the SGT system is delayed for the first 20 minutes while a negative pressure condition is being established in secondary containment. The basis for this 20-minute drawdown time is provided in the secondary containment drawdown discussion. Releases into the reactor building during the first 20 minutes are assumed to be directly exhausted to the environment as a ground level release with no filtration or hold-up. After 20 minutes, these releases are filtered by the SGT system.

Manual operator actions are credited for the actuation of drywell spray and the initiation of the SLC system. The manual actuation of drywell spray is assumed to occur within the first 15 minutes. This manual action is performed from the CR and is procedurally required. This is not a new manual action and the timing is bounded by the current licensing basis. Drywell spray is credited for scrubbing the primary containment atmosphere for the purpose of removing radioactive particulates and elemental iodine. The activity removed is assumed to be washed into the suppression pool. Credit for drywell spray for particulate removal is assumed for the time period of 15 minutes through 24 hours.

The manual injection of boron via the SLC system is credited for suppression pool pH control. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. This use of SLC is consistent with several other BWR submittals using AST. This is a new design basis requirement for SLC at Columbia. No hardware changes are necessary to implement this new requirement. The initiation of SLC is performed from the CR and is not a new manual action. New procedural guidance is required to address reliance on SLC for pH control. The appropriate procedural guidance will be established during the implementation of the LAR. (See section 4.8.1 for additional information on the SLC system and the justification for the use of SLC in this application.)

The main steam lines are seismically qualified up to the turbine stop valves. However, for conservatism, only the main steam line piping between the two MSIVs is credited for natural deposition (plateout). Additionally, to accommodate a postulated single failure of an MSIV to close, credit is taken for only three of the four steam lines. For the three credited lines, natural deposition was calculated according to AEB-98-03 (Reference 28) and a modified Bixler approach for gaseous iodine removal. For conservatism, the Bixler model was modified by adopting the AEB-98-03 well-mixed flow expression for gaseous iodine removal.

The assumed leakage rate from the primary containment and leakage via the MSIVs are reduced by a factor of two after 24 hours into the event. The reduction of this leakage rate is based on the ability of drywell spray to substantially reduce containment pressure within the first 24 hours of the event (see Assumption 2 of LOCA calculation in Attachment 5). Credit for reduction of primary containment leakage is consistent with the guidance in RG 1.183.

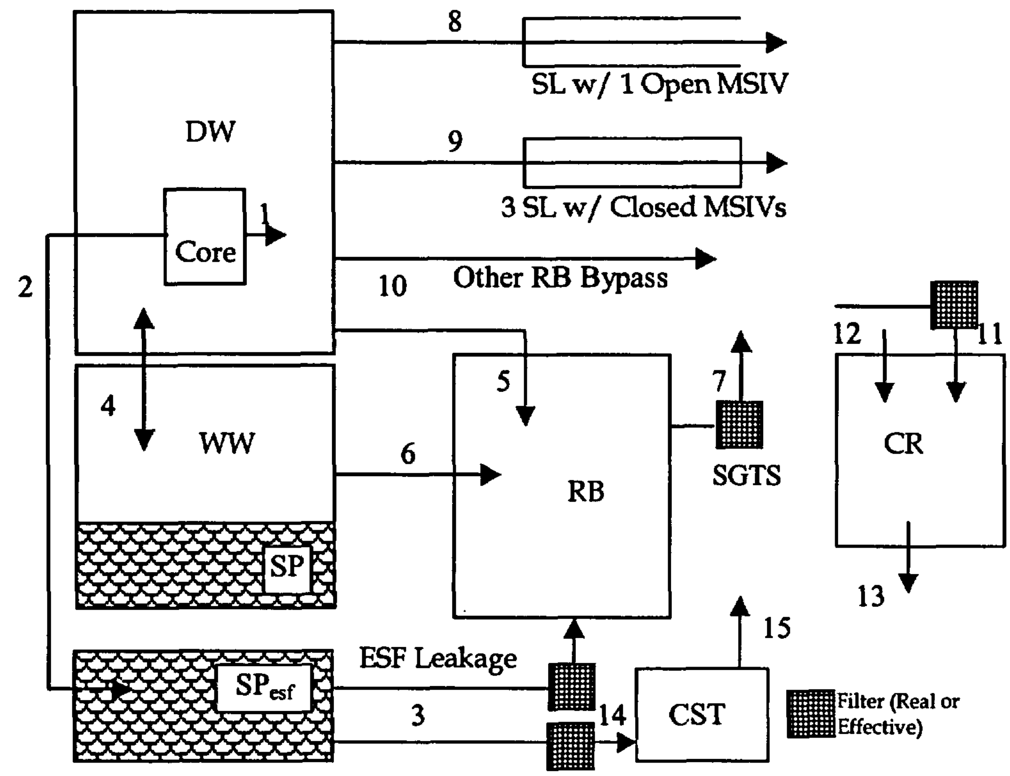
4.4.4 Radiological Transport Modeling

The radiological release model developed to calculate LOCA doses is shown in Figure 4.4-1. This model consists of seven control volumes.

CORE	Damaged core and reactor cooling system
DW	Drywell portion of the primary containment
WW	Wetwell portion of the primary containment
SP	Suppression pool
SP_{esf}	Suppression $Pool_{esf}$ (solely for modeling ESF leakage)
RB	Reactor building or secondary containment
CST	Condensate storage tanks
CR	Control room

The CST volume was included for the purpose of determining the significance of this source. Various junctions (flow paths) are modeled between and from the volumes. These junctions are associated with volumetric flows that determine the rate at which radioactivity is exchanged between the control volumes. In addition, removal processes such as deposition in pipes and filtration are modeled within and between the control volumes, as appropriate.

Figure 4.4-1
Release Model



A discussion of the pertinent aspects of these volumes and junctions is provided below.

Primary Containment – includes the Core, Drywell, Wetwell and Suppression Pool

The core volume is used to model the release of radioactivity to the drywell (Path 1) and to the suppression $pool_{esf}$ (Path 2) in parallel. Total release fractions were assumed to go through Path 1. To properly address ESF leakage, the total iodine release fractions were also assumed to go the suppression $pool_{esf}$. The release of iodines to the suppression $pool_{esf}$ was conservatively assumed to occur within the first two hours. The elemental and organic iodine released to the suppression $pool_{esf}$ was doubled to meet the relative ratio and percentage specified in RG 1.183 for ESF leakage.

Consistent with RG 1.183, the containment spray system was credited for a reduction in containment airborne activity. Credit was taken for drywell spray, relative to scrubbing the drywell atmosphere, for the first 24 hours of the event. Drywell spray initiation was assumed to occur at 15 minutes into the event. The crediting of drywell spray initiation in 15 minutes is reasonable relative to the FSAR analysis for ECCS performance and containment pressure response that assumes drywell spray initiation in 10 minutes. Reasonable assurance of the timeliness of this action is provided by two separate currently existing procedures.

- The emergency operating procedures (EOPs) direct the operator to initiate drywell sprays for containment pressure control if the drywell pressure exceeds 12 psig. The peak containment pressure for a design basis LOCA analysis would rapidly exceed this threshold.
- The Severe Accident Guidelines (SAGs) direct the operators to initiate drywell spray at a radiation level of greater than 14,000 rads/hour in the drywell. The AST LOCA calculation shows the radiation level in the drywell would exceed this threshold in a few minutes after the start of the gap release.

The drywell and wetwell are connected by downcomers and vacuum breakers, which allow steam relayed from the reactor to the drywell to flow to the suppression pool. Non-condensables could then collect in the wetwell gas space above the pool. When the drywell pressure is reduced by condensation (principally due to spray operation), a portion of these non-condensables will return to the drywell (Path 4). The suppression pool scrubbing of activity carried to the suppression pool by this process is not credited in this analysis.

Guidance from RG 1.183 and SRP Section 6.5.2 Revisions 1 and 2 was used to calculate removal rates. No credit for natural deposition in the drywell is taken, even when the sprays are not operating. The calculated removal rates are listed in Table 4.4-6.

Table 4.4-6 Aerosol Drywell Spray Removal Rates	
Time Frame	DW Spray Removal Rate (1/hr)
0 – 0.25 hr	0
0.25 – 2.44 hr	6.20
2.44 – 24 hr	0.62
24 – 720 hr	0

Leakage from the containment was modeled as 0.50% of the combined drywell (Path 5) and wetwell (Path 6) volumes per day. Prior to the completion of the secondary containment drawdown, this leakage was released directly to the environment. After drawdown (20 minutes), this leakage was filtered by the SGT system (Path 7) prior to being released to the environment. The assumed leakage rate from the primary containment was reduced by a factor of two after 24 hours into the event. Credit for this reduction of primary containment leakage is consistent with the guidance in RG 1.183.

Reactor Building Volume

The reactor building has a large free volume, but it was not credited for holdup. For modeling purposes, the SGT system was assumed to have a flow rate of 5000 cfm. During the drawdown period (i.e., the first 20 minutes), the secondary containment function was assumed to be completely bypassed. The SGT system filter efficiency for all forms of iodine and for particulates is 99%. A filter bypass of 50 cfm was also assumed. This reduces the filter efficiency to an effective value of 98%.

Secondary Containment Bypass Leakages

Two sources of leakage from the primary containment bypass secondary containment. These are MSIV leakage (Paths 8 and 9) and miscellaneous leakages (Path 10). From a dose contribution perspective, MSIV leakage is the more significant source of secondary containment bypass leakage. A new limit of 16 scfh per valve, or 64 scfh for four steam lines, at a test pressure of 39.7 psia (25 psig) is proposed in the TS change submitted with the LAR. This limit translates to an 8.3 cfh volumetric flow rate per penetration at the accident conditions of 52.1 psia (37.4 psig) maximum drywell pressure and a temperature of 283°F. Credit for natural deposition within the main steam lines was taken. To accommodate a postulated single failure of an MSIV to close, credit for natural deposition was taken for only three of the four steam lines. MSIV leakage was reduced by a factor of two at 24 hours.

The second source of bypass leakage, miscellaneous leakage paths, was assumed to equal the proposed TS limit of 0.04% primary containment volume per day at peak accident pressure. The supporting LOCA analysis was based on this limit for the first 24 hours. Consistent with the treatment of MSIV leakage, this leakage value was reduced by a factor of two at 24 hours.

No credit was taken for the main steam line leakage control system. The operability requirements for this system are being removed as part of the proposed TS changes.

<u>ESF Leakage</u>

Two sources of potential ESF leakage (Path 3) were included in the release model. The first is ESF system leakage directly into secondary containment. The current design basis assumes a value of one gpm. Consistent with RG 1.183, this value was increased by a factor of two. Leakage was assumed to start at t = 15 minutes after the event.

The second source of potential ESF leakage is into the CST. During the operation of high pressure core spray (HPCS) or reactor core isolation cooling (RCIC) systems aligned to the suppression pool, radiological impact of leakage into the CST through the CST suctions and test returns has been evaluated (Path 14). The contribution of the CST to the calculated doses (Path 15) is not significant and is not included in the dose results reported at the end of this section.

<u>Control Room</u>

The CR volume models the intake of activity from the environment for the purpose of calculating the dose to the control room operators. For the licensing basis case, one CREF train was assumed to fail at time zero leaving one train operating at 800 cfm (Path 11). The assumed CREF filter efficiencies were 95% for the gaseous iodine species and 99% for the particulates. The unfiltered inleakage for the single CREF train scenario was 50 cfm (Path 12). The CR exit flow rate (Path 13) is the sum of filtered and unfiltered incoming flow rates (Paths 11 and 12).

From a single failure perspective, the assumption of a single failure in the CREF system was conservative since this failure was analyzed as occurring simultaneously with the postulated single failure of an MSIV to close. The dose consequences associated with a single failure of a MSIV to close bound the consequences associated with a single failure of the CREF and the two failures are independent. Nonetheless, for conservatism, the mitigation of the LOCA with credit for only one CREF train is presented as the licensing basis case.

Two additional cases were evaluated. In these cases, both CREF trains were assumed to start as designed. In the first case, the CR operator was assumed to secure one of the two trains, eight hours after the start of the accident. In the second case, both trains were assumed to operate for the 30-day duration of the accident. The two-train filtered intake flow rate of 1300 cfm and an unfiltered inleakage of 75 cfm were used for these cases. The CR dose calculated for both of these scenarios is bounded by the single train licensing basis case discussed above. Securing a CREF train (when two trains are in operation following a design basis LOCA) before 8 hours could increase the dose to the operator. To preclude this undesirable operator action, the appropriate plant procedure(s) will be revised to prohibit the securing of a CREF train within the first 10 hours of the design basis LOCA.

<u>Summary of Release Model</u>

The general assumptions are:

- No credit for MSLC,
- Credit for spray removal in the drywell,
- No credit for natural deposition in containment,
- 0.50% volume per day primary containment leakage to the reactor building. This leakage rate is reduced by a factor of two at 24 hours,
- 0.04% volume per day primary containment leakage bypassing the reactor building. This leakage rate is reduced by a factor of two at 24 hours,

- MSIV leakage based on the TS leakage limit. Credit is taken for aerosol and iodine deposition in the three intact steam lines. MSIV leakage is reduced by a factor of two at 24 hours,
- 2 gpm of ESF leakage into secondary containment,
- Secondary containment drawdown time of 20 minutes,
- SGT system flow of 5000 cfm, with 50 cfm bypassing the filters,
- SGT filters: 99% efficient for all species except noble gases,
- No credit for holdup in the secondary containment, and
- CR air intake filters: 95% efficient for gaseous iodine, 99% for particulates.

4.4.5 Results - Control Room Operator Dose

The STARDOSE computer code (Reference 29) was used to determine the CR operator dose. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD (Reference 30), were used in STARDOSE. Table 4.4-7 shows the proposed licensing basis dose limit compared to the regulatory limit. Table 4.4-8 shows the non-license basis scenario results.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.4-7
LOCA CR Operator Dose
Licensing Basis Case

Scenario	TEDE	Regulatory Limit (TEDE)
Single failure of one CREF train	3.5 rem	5 rem

Table 4.4-8 LOCA CR Operator Dose Non-Licensing Basis Scenarios		
Scenario	**TEDE**	**Regulatory Limit (TEDE)**
Both CREF trains start and run for 30 days	3.2 rem	5 rem
Both CREF trains start and one is manually secured at eight hours	3.4 rem	5 rem

Sensitivity calculations were performed to evaluate the significance of dose contributions from ESF leakage to the CST and also for shine from the CREF filters. These calculations are included in the AST LOCA analysis in Attachment 5. Dose contribution from the CST is negligible, increasing CR operator dose by less than 1%. Dose contribution due to CREF filter shine is also negligible, approximately 1%.

Consistent with RG 1.183 Appendix A, item 4.2.1, a separate calculation was performed to assess the CR operator dose from shine dose from the reactor building, primary containment, and the plume outside the CR. These contributions were also shown to be negligible, less than .03% (Reference 31).

4.4.6 Results - Offsite Doses

The STARDOSE computer code was used to determine the offsite dose. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD, were used in STARDOSE. Table 4.4-9 shows the proposed licensing basis dose limit compared to the regulatory limit.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.4-9 LOCA Offsite Doses		
	TEDE	Regulatory Limit (TEDE)
EAB Dose*	4.1 rem	25 rem
LPZ Dose	4.0 rem	25 rem

* The EAB dose represents the maximum 2-hour TEDE over the accident period.

A sensitivity calculation was performed to evaluate the significance of the dose contribution from ESF leakage to the CST. The calculation is included in the AST LOCA analysis in Attachment 5. The dose contribution from the CST is negligible. The impact to the 30-day LPZ dose is less than 2%.

4.4.7 Conclusion

The LOCA CR operator dose is below the 5 rem TEDE regulatory limit and the offsite doses are well below the 25 rem TEDE regulatory limit.

4.5 Main Steam Line Break

4.5.1 Introduction and Background

The postulated MSLB accident assumes a double-ended break of one main steam line outside the primary containment. The assumed displacement of the pipe ends permits a maximum blowdown rate. The mass of coolant released is the amount in the steam line and connecting lines at the time of the break plus the amount passing through the MSIVs prior to closure (6 seconds). A total of 130,000 lbm of blowdown is released as documented in the current licensing basis. The quantity of blowdown is not affected by the application of the AST methodology to this event.

The release of steam to the environment resulting from the MSLB is assumed to be an instantaneous ground level puff. The methodology used to establish the puff transit time and the normalized concentration as a function of distance traveled is consistent with RG 1.194. The initial volume of the puff is established by the amount of steam released by the MSLB and by flashing a portion of the entrained liquid. The volume of the puff was calculated to be approximately 5.9E4 m^3.

The puff centerline passes directly over the local CR air intake. No credit is taken for expansion in the vertical (z) direction in performing the normalized concentration integration.

Two source term cases for the released coolant are considered. One is a pre-accident spike case of 4 μCi/gm dose equivalent (DE) I-131 and the second is a maximum equilibrium case of 0.2 μCi/gm DE I-131. These source term assumptions are consistent with RG 1.183.

The key parameters used in the MSLB analyses are shown in Table 4.5-1.

Table 4.5-1 Key Parameters for AST MSLB Analysis	
Columbia Design Input Parameter	**Parameter Value**
MSIV closure time	6 sec
Liquid release from MSLB	105,000 lbm
Steam release from MSLB	25,000 lbm
Reactor coolant system pressure	1060 psia
Reactor coolant system temperature	552°F
Distance from MSLB release point (assumed to be the closest blowout panel, panel A) to local CR intake	200' (61 m)
Puff volume	5.9E+04 m^3
Plume transit velocity	1 m/s
Maximum equilibrium iodine concentration	0.2 μCi/gm DE I-131
Pre-accident spike iodine concentration	4.0 μCi/gm DE I-131
Radioactivity release rate to environment	Instantaneous
Volume of CR	214,000 ft^3
CR occupancy factor	1
CR normal, unfiltered makeup flow	1100 cfm
Breathing Rate (both CR and offsite)	3.5E-4 m^3/sec
χ/Q, CR (puff model)	8.19E-4 sec/m^3
χ/Q, EAB	1.81E-4 sec/m^3
χ/Q, LPZ	4.95E-5 sec/m^3
Dose conversion factor for I-131 CEDE	32,893 rem/Ci

4.5.2 Source Term

The fission product inventory available for release was based on the maximum equilibrium reactor coolant DE I-131 concentration of 0.2 μCi/gm. This is the limit specified in TS LCO 3.4.8. In addition to the maximum equilibrium case, RG 1.183 requires a pre-accident iodine spiking case. To account for iodine spiking, the equilibrium level of DE I-131 was increased by a factor of 20 to achieve a spiking concentration of 4.0 μCi/gm. No fuel damage was postulated for the MSLB.

The methodology and assumptions used to calculate the total number of curies in the source term are consistent with RG 1.183 and the current licensing basis. The activity (in the terms of DE I-131) in the mass of the initial liquid blowdown was assumed to be released to the atmosphere instantaneously, as a ground level release, and no credit was taken for plateout, holdup, or dilution within facility buildings. For example, the DE I-131 total activity release for the iodine spiking case is 4 μCi/gm x 105,000 lbm (mass of the initial liquid blowdown) x 454 gm/lbm /1E6 μCi/Ci = 191 Ci.

4.5.3 Mitigation

The only mitigative action credited for the MSLB event was the termination of the release upon the automatic closure of the MSIVs. The MSIV closure time was assumed as 6 seconds. The 6 second closure time is consistent with the current licensing basis and is supported by TS SR 3.6.1.3.6. This surveillance requires the performance of periodic stroke time tests with an acceptance criteria of greater than or equal to 3 seconds and less than or equal to 5 seconds.

The CR ventilation was assumed to remain in the normal mode. The local air intake is used for analyzing dispersion. There is no accident signal credited to start emergency CR ventilation. No credit was taken for operator actions. The MSIV isolation actuates on a high flow signal. The CR ventilation normal intake flow was unfiltered.

4.5.4 Radiological Transport Modeling

The release of steam resulting from the MSLB (through blowout panels in the steam tunnel) was assumed to be an instantaneous ground level puff. The release point was assumed to be blowout panel A. It was assumed that the plume translates directly to the local CR intake which is closest to the assumed release location.

During normal operations, flow is through the local CR intake combined with flow from the remote intakes. The analysis assumed all 1100 cfm (maximum normal) of unfiltered supply air enters through the local intake. Other release locations and plume paths, such as a release via the turbine building, were considered in the MSLB calculation (Attachment 5). These sensitivity evaluations concluded the blowout panel A release point was bounding.

The RG 1.194 methodology was used to establish the puff transit time, normalized concentration as a function of distance traveled in the downwind or "x" direction, and the time-integrated normalized centerline concentration. Equation 10 of RG 1.194 was used to calculate the CR χ/Q. Puff initial volume was established by the amount of steam released by the MSLB and by the flashing of a portion of the entrained liquid. The puff from the steam release (including the flashed steam) was assumed to be released at ground level with an initial volume corresponding to standard atmospheric conditions. No buoyancy was considered.

All the activity in the liquid was assumed to be released into the puff. The time required for the plume to transit to the local CR air intake was based on the plume moving with a horizontal velocity of 1 m/s. The puff centerline is assumed to pass directly over the local CR air intake. No credit is taken for expansion in the vertical "z" direction in performing the normalized concentration integration.

4.5.5 Results - Control Room Operator Dose

The STARDOSE computer code was used to determine the CR operator doses and are shown in Table 4.5-2.

Table 4.5-2
MSLB CR Operator Doses

Source Term Case	**TEDE**	**Regulatory Limit (TEDE)**
Dose with maximum equilibrium iodine	0.1 rem	5 rem
Dose with pre-accident iodine spiking	1.8 rem	5 rem

4.5.6 Results - Offsite Doses

The offsite doses were calculated manually using the formula:

Dose (rem) = [Activity Release (Ci)] x [χ/Q (s/m^3)] x [Breathing Rate (m^3/s)] x [Dose Conversion Factor (rem/Ci)]

The offsite dose calculation assumes a direct unfiltered release to the environment; but because of the greater distances to the EAB and LPZ boundary, the dispersed release is assumed to be a continuous plume, modeled with PAVAN. Plume dilution due to buoyancy is not credited.

Resulting offsite doses are shown in Table 4.5-3 and Table 4.5-4.

Table 4.5-3
MSLB Offsite Doses
(Doses with maximum equilibrium iodine)

	TEDE	Regulatory Limit (TEDE)
EAB Dose	2.0E-2 rem	2.5 rem
LPZ Dose	5.5E-3 rem	2.5 rem

Table 4.5-4
MSLB Offsite Doses
(Doses with pre-accident iodine spiking)

	TEDE	Regulatory Limit (TEDE)
EAB Dose	0.40 rem	25 rem
LPZ Dose	0.11 rem	25 rem

4.5.7 Conclusions

The MSLB CR operator dose for the maximum equilibrium case is a small fraction of the 5 rem TEDE regulatory limit. The dose for the pre-accident iodine spike case is also well below the 5 rem TEDE regulatory limit.

The MSLB offsite doses for the maximum equilibrium case are a small fraction of the 2.5 rem TEDE regulatory limit. The dose for the pre-accident iodine spike case is a small fraction of the 25 rem TEDE regulatory limit.

4.6 Control Rod Drop Accident

4.6.1 Introduction and Background

The postulated CRDA involves the rapid removal of a highest worth control rod resulting in a reactivity excursion. Core performance analyses show the energy deposition that results from this event is below the threshold postulated to damage fuel pellets or cladding. However, consistent with the current licensing basis, 1.8% of the fuel pins in the full core are postulated to be damaged, with melting occurring in 0.77% of the damaged rods (i.e., 0.014% of the core). A core average radial peaking factor of 1.7 was assumed in the analysis.

The CRDA is terminated by the average power range monitors (APRM) high flux scram signal. The activity released from the damaged fuel that reaches the turbine and condenser is released from the turbine building at ground level at a rate of 1% condenser volume per day for a period of 24 hours. No credit is taken for turbine building holdup or dilution.

The analysis assumptions for the transport, reduction, and release of the radioactive material from the fuel and the reactor coolant are consistent with RG 1.183.

The key parameters used in the CRDA analysis are shown in Table 4.6-1.

Table 4.6-1
Key Parameters for AST CRDA Analysis

Columbia Design Input Parameter	Parameter Value
Core power	3556 MWt
Radial peaking factor	1.7
Percentage of fuel pins damaged	1.8% (This percentage is equivalent to 850 pins out of 764 assemblies x 62 pins/assembly for 8x8 fuel.)
Fraction of damaged fuel pins melted	0.77%
Condenser leak rate	1.0% condenser volume per day for the 24 hour release period
Volume of condenser	144,000 ft^3
Volume of CR	214,000 ft^3
CR occupancy factor	0 - 24 hrs: 1 1 - 4 days: 0.6 4 - 30 days: 0.4
CR normal, unfiltered intake flow	1100 cfm
Breathing rate (both CR and EAB)	3.5E-4 m^3/sec
Breathing rate (LPZ)	0 - 8 hrs: 3.5E-4 m^3/sec 8 - 24 hrs: 1.8E-4 m^3/sec 1 - 30 days: 2.3E-4 m^3/sec
χ/Q, CR (turbine bldg to local CR air intake)	0 - 2 hrs: 4.70E-3 sec/m^3 2 - 8 hrs: 2.00E-3 sec/m^3 8 - 24 hrs: 1.03E-3 sec/m^3 1 - 4 days: 8.01E-4 sec/m^3 4 - 30 days: 7.69E-4 sec/m^3
χ/Q, EAB	0 - 2 hrs: 1.81E-4 sec/m^3
χ/Q, LPZ	0 - 8 hrs: 4.95E-5 sec/m^3 8 - 24 hrs: 3.69E-5 sec/m^3 1 - 4 days: 1.95E-5 sec/m^3 4 - 30 days: 7.81E-6 sec/m^3
Dose Conversion Factors	Based on FGR 11 and FGR 12 defaults for RADTRAD

4.6.2 Source Term

The source term used for the CRDA analysis was composed of releases from melted fuel and the gap activity from the fuel pins postulated to be damaged. This initial amount of activity was released into the reactor coolant at time zero. Activity in the reactor coolant available for release to the environment was calculated by applying transport fractions.

The core damage fractions and transport fractions for each radionuclide group shown in Table 4.6-2 are consistent with RG 1.183. The fraction of the core inventory available for release to the environment was calculated as follows:

- [Core fraction of fuel pins damaged (less the melted fuel fraction) x gap release fraction + core fraction of melted fuel x melted fuel release fraction] x fraction that reaches the condenser x fraction that is available for release to environment.

Table 4.6-2
Fraction of Core Activity Available for Leakage to the Environment

Radionuclide Group	Release Fraction from Gap to Coolant	Release Fraction from Melted Fuel to Coolant	Fraction of Activity That Reaches the Condenser	Fraction of Condenser Activity Avail. for Release to Environment	Total Activity Fraction Avail. for Leakage to Environment
Noble Gas	0.1	1.0	1.0	1.0	1.9E-03
Iodine	0.1	0.5	0.1	0.1	1.9E-05
Br*	0.05	0.3	0.01	0.01	9.3E-08
Cs, Rb	0.12	0.25	0.01	0.01	2.2E-07
Te Group	0	0.05	0.01	0.01	6.9E-10
Ba, Sr	0	0.02	0.01	0.01	2.8E-10
Noble Metals	0	0.0025	0.01	0.01	3.5E-11
Ce Group	0	0.0005	0.01	0.01	6.9E-12
La Group	0	0.0002	0.01	0.01	2.8E-12

* Bromine is listed for consistency with RG 1.183 for Halogens, but is not included in the dose analysis.

The iodine species released to the reactor coolant are assumed to be 95% aerosol, 4.85% elemental, and 0.15% organic. The iodine species released from the condenser to the environment are 97% elemental iodine and 3% organic iodine. To properly model this release speciation in STARDOSE, the proportions of 97% elemental and 3% organic were used at the source.

The condenser is assumed to be leaking to the environment at a rate of 1% condenser volume per day during the first 24 hours, at which time the leakage is assumed to terminate.

4.6.3 Mitigation

The CRDA is terminated by the APRM high flux scram signal. Partitioning of the initial activity released during its transport from the reactor coolant system (RCS) to the condenser and ultimately to the environment was credited. Radioactive decay during the holdup in the turbine and condenser was also credited.

No other mitigation of the radiological release was credited. No credit for dilution or holdup in the turbine building was assumed. The CR ventilation was conservatively assumed to remain in its normal mode. There was no accident signal credited to start emergency CR ventilation. No credit was taken for operator actions. CR ventilation normal intake flow was unfiltered.

4.6.4 Radiological Transport Modeling

The radiological release model for the CRDA was developed consistent with RG 1.183. A ground level release was modeled from the turbine building at a rate of 1% condenser volume per day over a period of 24 hours.

During normal operations, flow is through the local CR intake combined with flow from the remote intakes. The intake of the released radionuclides into the CR is based on a volumetric flow rate of 1100 cfm of unfiltered air through only the local intake. This assumption is conservative, because no manual action for CR isolation was credited for the entire 24-hour period.

4.6.5 Results – Control Room Operator Dose

The STARDOSE computer code was used to determine the CR operator dose. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD, were used in STARDOSE. Table 4.6-3 shows the proposed licensing basis dose limit compared to the regulatory limit.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.6-3 CRDA CR Operator Dose		
	TEDE	**Regulatory Limit (TEDE)**
CR operator dose	0.7 rem	5 rem

4.6.6 Results – Offsite Doses

The STARDOSE computer code was used to determine the offsite dose. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD, were used in STARDOSE. Table 4.6-4 shows the proposed licensing basis dose limit compared to the regulatory limit.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.6-4 CRDA Offsite Doses		
	TEDE	**Regulatory Limit (TEDE)**
EAB dose	0.03 rem	6.3 rem
LPZ dose	0.03 rem	6.3 rem

4.6.7 Conclusions

The CRDA CR operator dose is well below the 5 rem TEDE regulatory limit and each offsite dose is a small fraction of the 6.3 rem TEDE regulatory limit.

4.7 Fuel Handling Accident

4.7.1 Introduction and Background

The postulated FHA (licensing base case) involves the drop of a fuel assembly in the reactor vessel cavity over the reactor core during refueling operations. At this location, the maximum drop (free fall distance) is approximately 34' and fuel pin damage is postulated to occur to both the dropped assembly and to some portion of those assemblies impacted in the reactor core.

The extent of damage is calculated based on the free fall distance and the resulting kinetic energy of the dropped assembly. In accordance with the current licensing basis, this drop is conservatively postulated to damage 250 fuel pins (based on a fuel assembly with an 8x8 fuel pin array).

The gap activity from the damaged pins is the radioactive source term for this event. A radial peaking factor of 1.7 is assumed in the analysis. A 24-hour decay time after plant shutdown is also assumed. This minimum decay time is assured by the proposed Decay Time TS.

An overall DF of 200 for the released iodines was assumed based on a minimum water depth of 23'. The nominal water depth (i.e., the distance from the top of the water above the vessel to the point of impact for the dropped assembly) for the postulated drop would be approximately 52' (well in excess of the credited 23').

The analysis assumed a ground level release from the reactor building over a 2-hour period. No credit was taken for secondary containment, the SGT system or the CREF system. The assumptions used in this analysis are consistent with RG 1.183.

Dropping a fuel assembly at other locations during fuel movement has also been considered. For a drop in the fuel transfer area (between the reactor vessel and the spent fuel pool) or over the spent fuel pool, the resulting maximum credible drop height would be significantly less than that assumed in the postulated FHA.

For a drop in the fuel transfer area (between the reactor vessel and the spent fuel pool) or over the spent fuel pool (see Figure 4.7-1), the postulated activity released would be substantially lower based on the following:

- The maximum credible drop height is 17". At a drop height of 17", the kinetic energy available to cause fuel damage is substantially reduced. The number of pins damaged in the design basis drop would bound the number of pins damaged in a drop elsewhere as the drop height is significantly greater in the licensing basis case.

- The TS minimum required water depth available over the point of fuel assembly impact is approximately 22', just 1' lower than the 23' upon which a DF of 200 is based. The difference in water height is approximately 1% for normal water level conditions (22' 9") and a maximum difference of approximately 4% for the minimum TS water level (22').

- The drop height of 17" is limited by procedural controls. In accordance with Licensee Controlled Specification 1.9.1, top of active fuel in an assembly must be maintained at least 7' 6" below the TS minimum required water level of 22'.

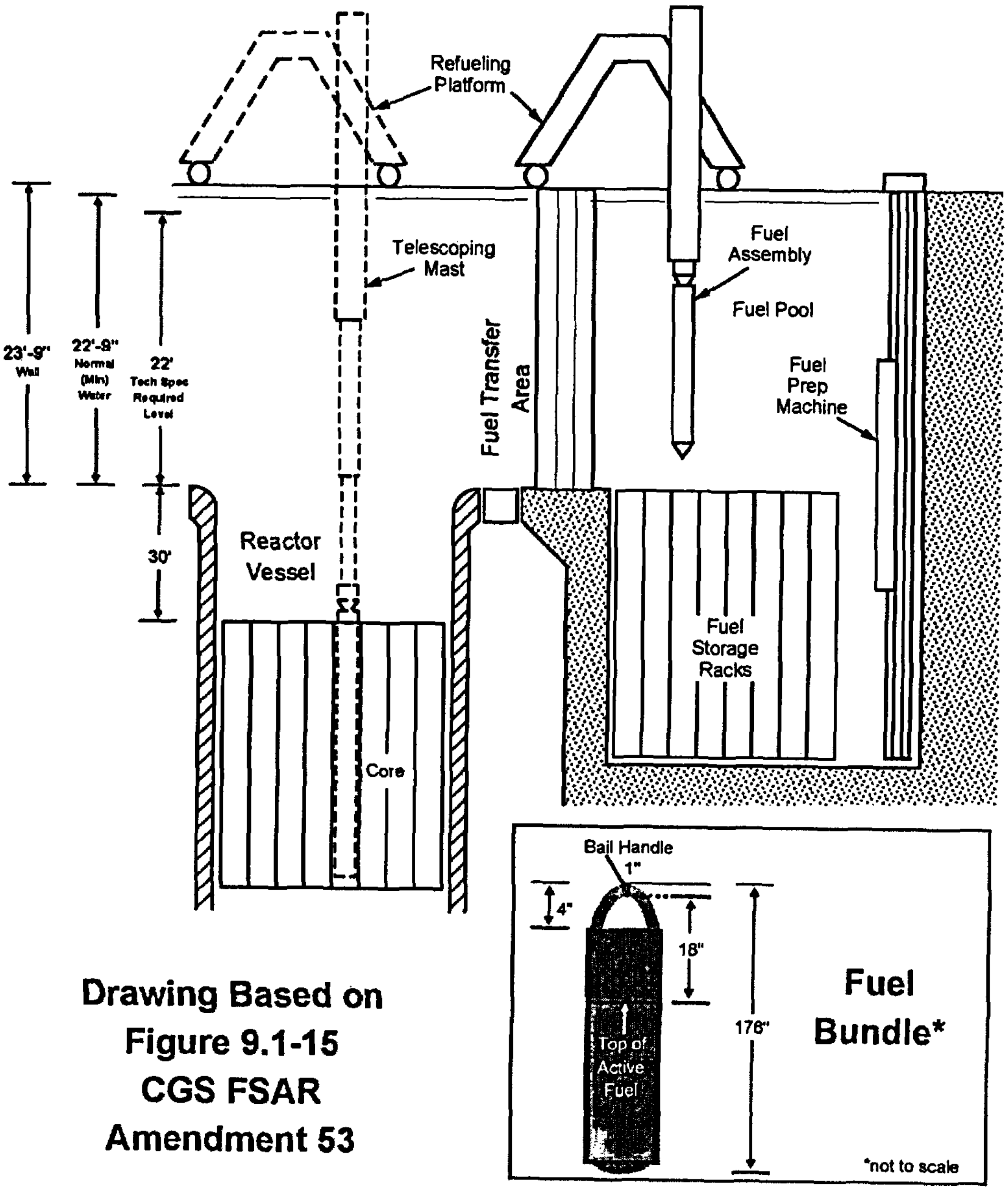
Based on the comparable water depth available for decontamination and the difference in the postulated drop distances, Energy Northwest concludes that the consequences of an FHA over the reactor cavity bound those for an FHA over the transfer area or over the spent fuel pool. This conclusion is consistent with the NRC staff conclusion for a similar configuration at the Fitzpatrick plant as documented in a recent Safety Evaluation Report (SER) (Reference 32).

The key parameters used in the FHA analysis are shown in Table 4.7-1.

Table 4.7-1
Key Parameters for AST FHA Analysis

Columbia Design Input Parameter	**Parameter Value**
Core power	3556 MWt
Peaking factor	1.7
Decay time	24 hours
Fraction of fuel damaged in drop	0.53% (based on 250 pins of an 8x8 array)
Water depth (licensing basis case)	> 23'
Overall Iodine DF	200
Radioactivity release rate to environment	Greater than 99% of the available activity released within 2 hours. A fractional release rate of 2.3 volumes per hour was used for modeling purposes.
Volume of CR	214,000 ft^3
CR occupancy factor	1 (first 24 hours after release)
CR normal, unfiltered intake flow	1100 cfm
Breathing Rate (both CR and offsite)	3.5E-4 m^3/sec
χ/Q, CR	8.69E-4 sec/m^3 (RB wall to local CR air intake)
χ/Q, EAB	1.81E-4 sec/m^3
χ/Q, LPZ	4.95E-5 sec/m^3
Dose conversion factors	Based on FGR 11 and FGR 12 defaults for RADTRAD

Figure 4.7-1
Fuel Handling Figure



Drawing Based on Figure 9.1-15 CGS FSAR Amendment 53

4.7.2 Source Term

The fission product inventory that constitutes the source term for this event was the gap activity in the 250 fuel pins (based on a fuel assembly with an 8x8 fuel pin array) assumed to be damaged as a result of the postulated design basis FHA. This number of fuel pins equates to 0.53% of the total number of fuel pins in the reactor core. Of this activity, all of the noble gases and only a fraction of the iodine were available for release (i.e., for the purpose of calculating radiological dose consequences) based on the scrubbing effect (i.e., DF) of the water above the dropped fuel assembly. Consistent with RG 1.183, an overall DF of 200 was credited for the various forms and isotopes of iodine and an infinite DF was credited for the remaining particulate forms of the radionuclides contained in the gap activity. No DF credit was taken for the noble gas constituents of the gap activity.

The fission product inventory assumed to be gap activity was based on the fractions (shown in Table 4.7-2) of the core fission product inventory. These fractions were taken from Table 3 of RG 1.183. After applying these fractions to determine the quantity of radioactive nuclides in the gap, a decay time of 24 hours is applied.

Table 4.7-2
FHA Analysis Gap Activity
(Fraction of Fission Product Inventory)

Radionuclide Group	Fraction of Core Inventory
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

4.7.3 Mitigation

Decontamination of the gap activity as it rises (bubbles) to the surface through the water above the dropped assembly in the reactor vessel was credited. No other mitigation of the radiological release was credited. The proposed TS changes delete the operability requirements for secondary containment, SGT system and the CREF system during fuel handling or core alterations. This analysis demonstrates acceptable radiological consequences are achievable without crediting these systems.

During this event, the CR ventilation remains in its normal mode. The local air intake was used for analyzing dispersion. No accident signal was credited to start emergency CR ventilation. No credit was taken for operator actions. The CR ventilation normal intake flow was unfiltered.

4.7.4 Radiological Transport Modeling

The radiological release modeled in this analysis is consistent with RG 1.183.

The release of the gap activity from the damaged pins is modeled to occur instantaneously. For modeling purposes, a fractional release rate of 2.3 volumes per hour was utilized to ensure that at least 99% of the activity was released from the reactor building during the first 2 hours.

The CR χ/Q for the worst-case release path from the reactor building was the reactor building wall release point to the local CR air intake. Since secondary containment operability was not required for fuel handling activities, various potential pathways are possible and were considered (e.g., reactor building stack, reactor building wall, vehicle air lock doors). For conservatism, the most limiting pathway was selected.

4.7.5 Results – Control Room Operator Dose

The STARDOSE computer code was used to determine the CR operator dose. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD, were used in STARDOSE. Table 4.7-3 shows the proposed licensing basis dose limit compared to the regulatory limit.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.7-3 FHA CR Operator Dose		
	TEDE	**Regulatory Limit (TEDE)**
CR operator dose	3.7 rem	5 rem

4.7.6 Results – Offsite Doses

The STARDOSE computer code was used to determine the offsite doses. Dose Conversion Factors (DCFs) from the Federal Guidance Report 11 and 12, defaults for RADTRAD, were used in STARDOSE. Table 4.7-2 shows the proposed licensing basis dose limit compared to the regulatory limit.

A confirmatory analysis using RADTRAD for the licensing basis case was performed. The results of this confirmatory analysis showed good agreement with the STARDOSE results.

Table 4.7-4 FHA Offsite Doses		
	TEDE	**Regulatory Limit (TEDE)**
EAB Dose	1.0 rem	6.3 rem
LPZ Dose	0.3 rem	6.3 rem

4.7.7 Conclusions

The FHA CR operator dose is below the 5 rem TEDE regulatory limit and each offsite dose is well below the 6.3 rem TEDE regulatory limit.

4.8 Miscellaneous Issues

4.8.1 Use of Standby Liquid Control

This section provides the basis for crediting boron injection from the SLC system for suppression pool pH control. The maintenance of a suppression pool pH level above 7.0 is important to prevent re-evolution of iodine from the suppression pool water. This use of

SLC is consistent with several other BWR submittals using AST methods. The SLC system is categorized in the FSAR as a special safety system with a design function to mitigate the Anticipated Transient Without Scram (ATWS) event per 10 CFR 50.62.

No hardware changes are necessary to use SLC in this new functional mode. However, a change to the SLC TS is proposed in this LAR to add MODE 3 to the applicability statement. This proposed TS change supports the SLC function as credited in the AST LOCA analysis.

The SLC system, shown in Figure 4.8-1, consists of a heated storage tank containing a low temperature sodium pentaborate decahydrate solution, two positive-displacement pumps connected in parallel, two motor operated suction valves, two explosive actuated discharge valves, a test tank with its network of injection and recirculation pipes, and the necessary piping valves and instrumentation needed to inject the boron solution into the reactor vessel. The SLC system is manually initiated from the CR. Both SLC trains are initiated by redundant switches.

Upon initiation, the suction valves of both trains will open, the pumps will start, and both explosive actuated discharge valves open. This establishes a flow path for the boron solution from the storage tank into the reactor vessel. The boron solution discharges inside the shroud through the HPCS spray header. The positive displacement pumps are sized to inject the contents of the storage tank solution into the reactor in approximately 1 hour.

In February 2004, the NRC issued review guidelines (Reference 33) for assessing the acceptability of a BWR SLC system for pH control. These guidelines have been the basis of several recent Requests for Additional Information (RAIs). Energy Northwest has evaluated the SLC system against these guidelines. The following assessment is formatted with the guidelines in bold italic and the Energy Northwest response in standard text.

Based on the following response, Columbia satisfies the criteria for the acceptability of the SLC system for pH control.

1. ***The SLC system should be classified as ESF grade in accordance with 10 CFR 50.34(b) or as a safety-related system as defined in 10 CFR 50.2, and satisfy the regulatory requirements for such systems. There may be plants with an SLC system which is not classified as safety-related or as ESF grade. In such instances, the staff reviewer will determine whether the SLC system is***

comparable to a system classified as safety-related or ESF. A SLC system meeting items (a)-(e) below would result in its acceptance in support of a 10 CFR 50.67 request even if the system is not classified as safety-related or as ESF grade.

The SLC system is not classified as safety-related nor as ESF grade. The basis for SLC system meeting items (a) through (e), and therefore the acceptability of the SLC system for the AST pH control application, is provided below.

a) The SLC system should be provided with standby AC power supplemented by the emergency diesel generators.

The SLC system is provided with standby AC power supplemented by emergency diesel generators.

SLC has redundant electrical components requiring AC power to actuate for injection. Separate safety-related AC divisions, both backed by onsite emergency diesel generators, power their respective components.

b) The SLC system should be seismically qualified in accordance with RG 1.29 and Appendix A to 10 CFR Part 100.

The SLC system is seismically qualified from the storage tank (including the tank) to the injection point to the HPCS piping. Seismic qualification of the SLC system is in accordance with RG 1.29 and Appendix A to 10 CFR 100.

c) The SLC system should be incorporated into the plant's ASME Code ISI and IST Programs based upon the plant's code of record (10 CFR 50.55a).

The SLC system components and piping are included in the Columbia ISI and IST Programs. The only recorded failure has been an inboard check valve failure of leak tightness. This failure would not have prevented the system from meeting its new design function.

d) ***The SLC system should be incorporated into the plant's Maintenance Rule program consistent with 10 CFR 50.65.***

The SLC system has been included in the Maintenance Rule Program per 10 CFR 50.65 since implementation of the program at Columbia in July 1996.

e) ***The SLC system should meet 10 CFR 50.49 and Appendix A (GDC 4) to 10 CFR 50.***

The SLC system components have been qualified to operate in a post-LOCA environment as defined in the Columbia EQ program.

2. ***The licensee should have plant procedures for injecting the sodium pentaborate using the SLC system. This information would be reviewed by the appropriate technical review branch, as requested by the lead SPSB reviewer.***

(a) ***A review of the procedures may be appropriate if a reliability approach is taken (4(a) below) due to timing considerations for the injection of chemicals.***

Energy Northwest has taken a reliability approach and therefore, responses to items 2(a) through (f) are provided.

The operator determines the need for SLC system usage. Manual initiation of the SLC system is directed by the EOPs (inventory as an alternate injection path and ATWS usage) and the SAGs (reactivity and inventory control) that are safety-related.

The Technical Support Guidelines (TSGs) will be revised to add a second functional use of SLC for pH control. The TSGs will also reinforce the need to flood the vessel, if required, for communication of the boron solution to the suppression pool. The timing issue is discussed more completely in response to item 3 below.

(b) The SLC activation steps are placed in a safety-related plant procedure.

The AST LOCA analysis is based on low reactor water level without ECCS system injection to generate the source term. The EOP PPM 5.1.1 lists SLC as an alternate injection system. The inadequate core cooling resulting in the source term requires entry into SAG-1 that requires SLC injection. Both of these procedures are safety-related. A release of the AST magnitude would result in the containment radiation monitors reading high. The TSGs will be revised to require manual initiation of the SLC system, at a level of 14,000 R/hr, and to continue injection until the SLC tank low level alarm is received.

(c) The steps be activated by parameters that are symptoms of imminent or actual core damage.

The inability to maintain water level or potential loss of water level indication and the containment high radiation signal are signals of imminent and actual core damage, respectively.

(d) The instrumentation relied upon to provide this indication meets the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2.

The containment radiation instrumentation is required to be operable in accordance with the LCS and meets the quality requirements for a Type E variable as defined in RG 1.97 Tables 1 and 2. The water level instrumentation is required to be operable in accordance with TS and meets the quality requirements for a Type B variable as defined in RG 1.97 Tables 1 and 2.

(e) Personnel receive initial and periodic refresher training in the procedure.

Licensed operators receive initial and periodic training on procedure changes as part of their requalification training. In addition, Technical Support Center (TSC) Operations Managers will receive training on the TSG revisions as part of the implementation of the approved AST changes.

(f) ***Other plant procedures (e.g., ERGs/SAGs) that call for termination of SLC as a reactivity control measure are appropriately revised to enable SLC injection for pH control.***

The SAG-1 does not call for or provide any instruction for termination of SLC. In addition, the changes to the TSGs for high containment radiation will instruct the operators to inject until low tank signal is received.

3. ***A sufficient concentration and quantity of sodium pentaborate should be available for injection into the reactor vessel to control pH in the suppression pool. The source term analysis is tied to the plant's design basis accident, which is the large break LOCA, a break of a recirculation pipe. The licensee needs to demonstrate that within 24 hours there is adequate recirculation between the suppression pool and the reactor vessel through flow out the break to provide transport and mixing, consistent with the assumptions in the chemical analyses.***

Chemical pH Analysis

The pH calculation was prepared using the STARPH code (Reference 34). The analysis demonstrates one tank of boron solution contains sufficient boron to maintain suppression pool pH > 7.0 for 30 days and was based on minimum TS requirements for the SLC system. This calculation is provided in Attachments 5 and 6.

The initial phase of the design basis LOCA will release large amounts of fission products. Several of the fission product chemical forms are pH basic, most notably, CsOH. The CsOH is introduced immediately ensuring an initial pH of greater than 7.0. The pH calculation also shows that formation of acids HNO_3 from radiolysis of water and HCl from radiolysis of cable would require a minimum of 8 hours to reduce the suppression pool pH to less than 7.0 without the addition of a boron buffer.

With only one of the two SLC trains operating, the contents of the SLC tank can be injected into the vessel in approximately 2 hours. Therefore, the addition of a boron buffering solution to the suppression pool by the SLC system is adequate for controlling the suppression pool pH. An adequate amount of boron is provided by borated solution storage tank based on the minimum volume and concentration required by the SLC system TS.

<u>Transport and Mixing</u>

Columbia is a Mark II containment design with the drywell over the suppression pool and communication between them via 99 downcomers. Following a design basis LOCA, ECCS will inject to quickly reflood the RPV to the top of the jet pumps for a recirculation line break or to the steam lines for a main steam line break. Mixing will therefore start immediately upon injection of the boron.

Either division of low pressure ECCS can provide at least one containment spray and one low-pressure core injection or core spray system. Each division of ECCS minimum flow is approximately 13,000 gpm, which is approximately (13,000 x 0.134 ft^3/gal =) 1700 ft^3/min. The reactor vessel water volume is approximately 13,000 ft^3 and the suppression chamber free volume is approximately 140,000 ft^3. Therefore, one complete vessel plus suppression pool will recirculate in approximately 1.5 hours.

The SLC injection will complete in 4 hours, assuming 2 hours to initiate plus 2 hours to inject one tank capacity. Within the first 24 hours there would be approximately 13 recirculations of the suppression pool after the completion of the SLC injection assuring a well-mixed solution.

4. ***The SLC system should not be rendered incapable of performing its AST function due to a single failure of an active component. For this purpose the check valve is considered an active device for AST since the check valve must open to inject sodium pentaborate for suppression pool pH control.***

   If the SLC system can not be considered redundant with respect to its active components, this lack of redundancy may be offset if the licensee can satisfy (a) or (b) or (c) below:

   There are two in-series check valves in the SLC system injection line. With the exception of the failure of either of these check valves to open, the SLC system cannot be rendered incapable of performing its AST function due to a single active component failure. Energy Northwest has chosen to respond to 4 (a) below to offset the check valve single active failure concern. Therefore, no response is provided for items 4 (b) or 4 (c).

(a) ***Acceptable quality and reliability of the non-redundant active components and/or compensatory actions in the event of failure of the non-redundant active components.***

Under this approach, the licensee should provide the following information in justifying the lack of redundancy of active components in the SLC system:

(1) The licensee should identify the non-redundant active components in the SLC system and provide their make, manufacturer, and model number. The staff reviewer will compare this information with performance data for the component from industry data bases and other sources.

The non-redundant active components of the SLC system are lift check valves (two in series) located on the containment penetration for the SLC injection line. The type, manufacturer (make), and model number for the check valves are:

Check Valves	Type	Manufacturer	Model No.
SLC-V-6	lift check	Borg-Warner	76790-1
SLC-V-7	lift check	Borg-Warner	76790-1

(2) The licensee should provide the design-basis conditions for the component and the environmental and seismic conditions under which the component may be required to operate during a design-basis accident. Environmental conditions include design-basis pressure, temperature, relative humidity and radiation fields. The staff reviewer will compare the environmental and seismic conditions associated with the design-basis accident to the conditions for which the component was designed to determine whether the component is capable of performing its intended function.

The SLC system has been qualified for a post-LOCA environment for a period of 24 hours as a result of this new application. The system was originally designed and installed for seismic category I conditions.

(3) The licensee should indicate whether the component was purchased in accordance with Appendix B to 10 CFR Part 50. If the component was not purchased in accordance with Appendix B, the licensee should provide information on the quality standards under which it was purchased. For the latter situation, information on the component would be reviewed by the appropriate technical review branch responsible for the component, as requested by the lead SPSB reviewer.

The SLC injection line check valves were purchased as Quality Class I components. They are Quality Class I, ASME III Code Class I, and are maintained within the requirements of the Energy Northwest 10 CFR 50 Appendix B QA Program.

(4) The licensee should provide the performance history of the component both at the licensee's facility and in industry databases such as EPIX and NPRDS. The staff reviewer will use this information to evaluate the reliability of the component relative to other components used in safety-related applications.

Energy Northwest performed a search of the NPRDS database. No other plant was identified with SLC injection line check valves of the Borg-Warner Model 76790-1 type. However, these valves are used in various other applications at several commercial nuclear power plants. The industry data indicates that check valves less than 2 inches in diameter (SLC-V-6, -7 are 1 ½ inches diameter) are very reliable.

A failure summary report from the Institute of Nuclear Power Operations Equipment Performance and Information Exchange (EPIX 4.0) database shows no cases of the same model check valve failing to open.

The SLC system is included in the Maintenance Rule Program. System reliability and availability are tracked for any potential degradation of performance. The maintenance history of the Columbia SLC system has been reviewed as documented in the Maintenance Rule Program, the Check Valve Reliability Program, and the Corrective Action Program. These valves have shown reliable performance with no failures to open, consistent with the industry experience noted above.

Check valve SLC-V-7 is included in the Local Leak Rate Testing (LLRT) Program for its containment isolation function. Leak rate failures were identified in 1986, 1987, and 1992. These failures would not have prevented the SLC system from performing its injection function.

(5) The licensee should provide a description of its inspection and testing program including standards, frequency, and acceptance criteria. The staff reviewer will use this information to evaluate the licensee's activities to monitor the component's performance at the facility. The information on the component would be reviewed by the appropriate technical review branch responsible for the component, as requested by the lead SPSB reviewer.

The majority of the inspection and testing activities for the SLC system are driven by the TS 3.1.7 SR. These requirements include daily checks of the volume and temperature of the solution tank, monthly checks of system valve lineups, chemical analysis of the boron solution concentration.

Refueling outage tests include piping leak test in accordance with ASME III Code Class I piping, testing of safety/relief valves, testing of the inboard containment isolation check valve in accordance with the IST leak test program, and a full-flow injection into the RPV. The full-flow test verifies both check valves open. Acceptance criteria are provided in the TS or the IST program.

(6) The licensee should also indicate potential compensating actions that could be taken within an acceptable time period to address the failure of the component. An example of a compensating action might be the ability to jumper a switch in the control room to overcome its failure. The staff reviewer will consider the availability of compensating actions and the likelihood of successful injection of the sodium pentaborate where non-redundant active components fail to perform their intended functions.

The only identified single active failure identified for the Columbia SLC system is the failure of one of two lift check valves in series. Responses to items (1) through (5) above have demonstrated that these valves are highly reliable with no identified failures to open.

The Probabilistic Risk Assessment (PRA) database indicates the failure rate of these valves as 3.077E-04 per demand, which is comparable to a passive failure rate. Given the redundancy in the rest of the SLC active components and the reliability of the lift check valves, no compensating actions are proposed.

(b) An alternative success path for injecting chemicals into the suppression pool.

If the licensee chooses to address the SLC system's susceptibility to single failure by selecting an alternative injection path, the alternative path must be capable of performing the AST function noted above and all components which make up the alternative path should meet the same quality characteristics required of the SLC system (described in Items 1(a)-1(e), 2 and 3 above). When the staff determines that an alternative path is acceptable, the staff's safety evaluation should address the manner in which the SLC system and the alternative path met Items 1(a)-1(e), 2 and 3 above.

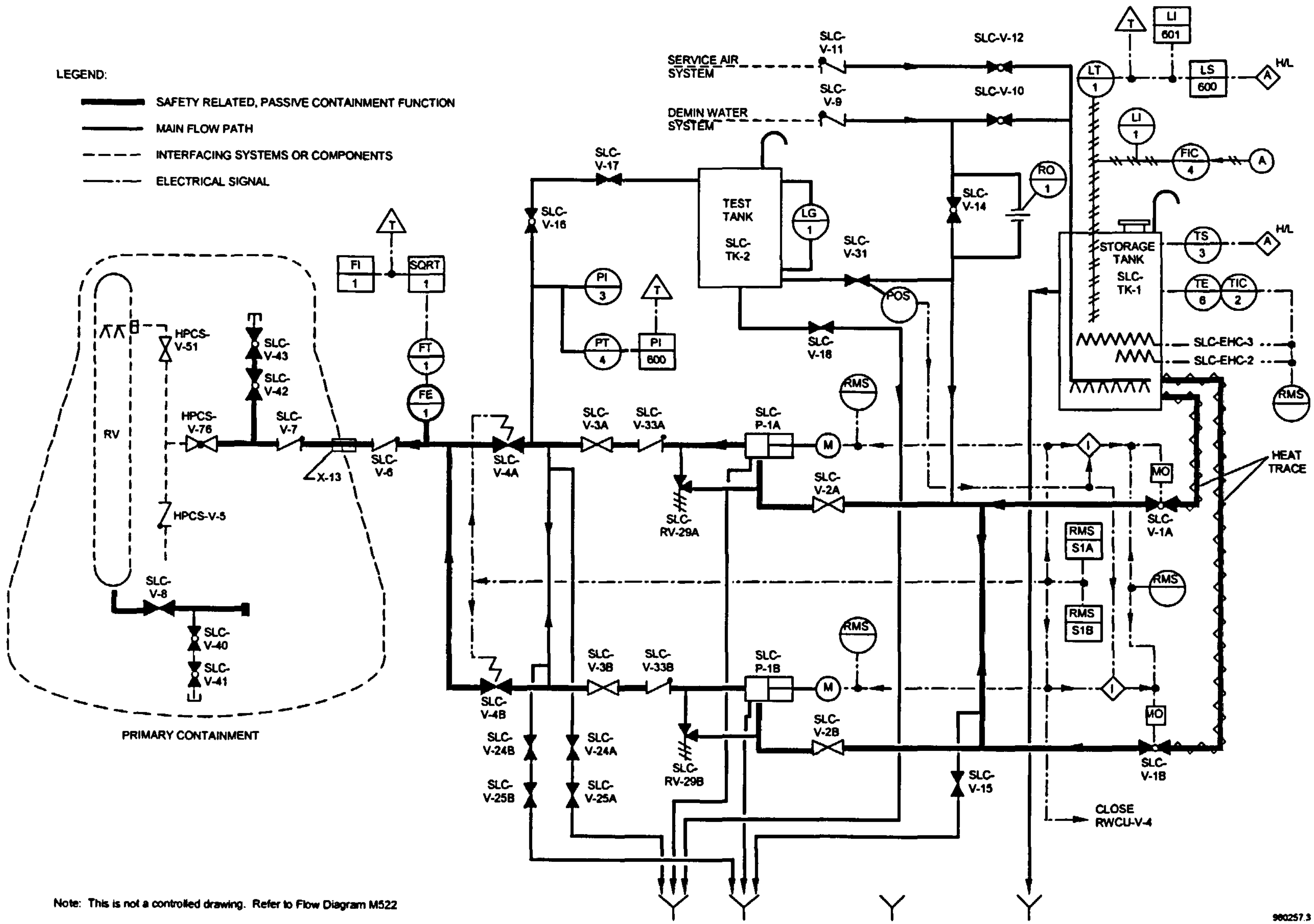
Columbia has responded to part (a) above, therefore, (b) is not applicable.

(c) ***10 CFR 50.67 and Appendix A, General Design Criterion (GDC) 19 doses are met even if pH is not controlled.***

The licensee may demonstrate, through dose calculations, that 10 CFR 50.67 and GDC 19 doses are met even if pH is not controlled. The re-evolution of iodine in the particulate form from the water in the suppression pool to the elemental form for airborne iodine must be incorporated into the calculation. The calculation may take credit for the mitigating capabilities of other equipment, for example the SGT system, if such equipment would be available. The staff will perform calculations to confirm the licensee's conclusions. If the acceptability of the facility's dose calculations was based on the utilization of certain ESF equipment, for example the SGT system, then the staff's safety evaluation should reflect this. Such a citation is necessary to assure that it is recognized and documented that there is a link between the particular ESF component's performance and the SLC system's susceptibility to single failure.

Columbia has responded to part (a) above, therefore, (c) is not applicable.

Figure 4.8-1
Simplified SLC System Flow Diagram



Note: This is not a controlled drawing. Refer to Flow Diagram M522

960257.3

4.8.2 Operator Actions

There are no new manual operator actions proposed as part of this LAR that are not already considered in the Columbia design basis and directed by established station procedures.

There are two operator actions assumed in the proposed AST dose consequence analyses: 1) initiating the SLC system for boron injection, and 2) initiating drywell sprays. The abnormal procedures, EOPs, and SAGs, as applicable, direct the operators to take these actions.

While the actions are the same, the additional reasons for the actions are: 1) drywell sprays are credited for analytical assumptions regarding drywell leakage and to reduce the radionuclide particulate concentration in the primary containment atmosphere; and 2) adding boron will maintain the suppression pool water pH above 7.0, precluding iodine re-evolution.

4.8.3 NUREG-0737, Item II.B.2

Equipment Qualification

The source term associated with environmental qualification of equipment will remain consistent with previous commitments under 10 CFR 50.49. As stated in the cover letter to this submittal, the Energy Northwest application to implement the AST methodology is requested with one exception. That exception is TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification.

TSC Habitability

The current licensing basis for the habitability of the TSC remains valid. As stated in the Columbia Emergency Plan:

> "The Columbia Generating Station Technical Support Center (TSC) is a structure attached to the Radwaste Building on the west side of the plant...The TSC ventilation HEPA filters and charcoal absorbers are the same type as those used for the CR System. The ventilation system local air intake will automatically switch (sic) to the CR remote air intake upon receipt of an isolation signal. [Clarification - The air intake structures (local and two remote intakes) are the

> same as those used for the Control Room System. The ventilation system air flow will automatically draw exclusively from the Control Room remote air intakes upon the isolation of the local intake at the receipt of an isolation signal.] Should the TSC become uninhabitable, the TSC Manager in consultation with radiation protection personnel will decide on an alternate TSC location and setup where the TSC functions can continue to be performed."

Based on the overall reduction in CR operator dose due to AST methodology, similarities in ventilation systems, and the ability to evacuate the TSC, an updated quantitative assessment of the TSC dose based on the AST source term was not performed.

<u>Emergency Operations Facility (EOF) Habitability</u>

The current licensing basis for the habitability of the EOF remains valid. As stated in the Columbia Emergency Plan:

> "The Emergency Operations Facility is a protected area in the Kootenai Building which has special shielding and ventilation to maintain habitability requirements. The ventilation system is designed to provide maximum habitability during an accidental radiological release. HEPA filters condition entering air during emergency conditions. Ion chambers are strategically located in the ventilation system to detect potential infiltration of contaminated air thus automatically allowing reconfiguration of airflows from replenishment to recirculation modes. The EOF is designed to ensure that the total dose to occupying personnel is less than the Environmental Protection Agency Protective Action Guide limit of 5 rem TEDE for the duration of the postulated accident. Shielding requirements were determined using source terms from BWR/PWR accident scenarios described in the WASH 1400 Reactor Safety Study. Calculations considered worst case meteorology and assumed a 0.75 miles distance from the plant to the Emergency Operations Facility.

> "The Alternate Emergency Operations Facility (EOF) is located approximately 10 miles south of the plant. This facility may be activated in the event of the primary EOF becoming uninhabitable, or inaccessible

> to offsite responders. The EOF Manager, in consultation with radiation protection personnel, will determine when to activate this facility and appropriate staffing levels."

Based on an overall reduction in doses due to AST methodology, the distance of the EOF from the plant, and the ability to evacuate the EOF, an updated quantitative assessment of the EOF dose based on the AST source term was not performed.

5.0 REGULATORY SAFETY ANALYSIS

10 CFR 50.92 Evaluation

Summary of Proposed Change

Energy Northwest is requesting an amendment to the Columbia Generating Station Operating License based on AST methodology. The alternative source term analyses were performed following the guidance of RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000, and Standard Review Plan Section 15.0.1, "Radiological Consequences Analyses Using Alternative Source Terms." (Revision 0)

The alternative source term analyses have been performed without crediting secondary containment or the control room emergency filtration system during fuel handling accidents. As such, the requested license amendment removes operability requirements during fuel handling and core alterations for: 1) secondary containment; 2) secondary containment isolation instrumentation; 3) the standby gas treatment system 4) the control room emergency filtration system and 5) supporting AC and DC power sources and distribution systems. These requested changes, combined with the addition of a Technical Specification on decay time, are consistent with Technical Specification Task Force (TSTF) Traveler 51.

The alternative source term analyses have been performed without crediting the main steam leakage control system following a loss of coolant accident. Therefore, a licensing basis change is requested to reflect the elimination of the main steam leakage control system Technical Specification. Additionally, relaxations to the main steam isolation valve leakage and secondary containment bypass leakage Technical Specifications are requested and justified by the application of the alternative source term.

This license amendment request resolves a Justification for Continued Operation regarding the establishment of secondary containment vacuum under design bases conditions that include adverse environmental conditions. This design basis nonconformance was reported to the staff in several licensee event reports (see Licensee Event Reports 88-023-00, 88-023-01, 89-040-00 and 89-040-01). A new license and design basis analysis on secondary containment drawdown, as credited for a loss of coolant accident, is provided with this amendment request. Accordingly, changes are requested to the secondary containment and standby gas treatment system Technical Specifications.

This request resolves a previously identified Unreviewed Safety Question pertaining to increased unfiltered control room leakage into the control room envelope (Licensee Event Reports 2000-006-00 and 2000-006-01). The application of alternative source term methodology demonstrates that increased unfiltered inleakage results in a control room operator dose below the regulatory limit. The increased inleakage limits bound the results of tracer gas testing.

5.1 No Significant Hazards Consideration Determination

The standards used to arrive at a determination that an amendment request does not involve a significant hazard are included in 10 CFR 50.92. Energy Northwest has evaluated the requested change to the Technical Specifications and licensing and design bases using the criteria established in 10 CFR 50.92(c) and has determined that it involves no significant hazards consideration as described as follows:

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

   Response: No.

   The alternative source term does not affect the design or operation of the facility in a manner that would impact the probability of an accident previously evaluated. Assumed performance requirements of the system structures and components are within existing design capability. The manner in which the systems are required to operate has not changed.

   Once the occurrence of an accident has been postulated, the new source term is an input to evaluate the consequences. The implementation of the alternative source term methodology has been evaluated in revisions to the analyses of the following limiting design basis accidents at Columbia Generating Station:

   - Control Rod Drop Accident
   - Fuel Handling Accident
   - Main Steam Line Break Accident
   - Loss of Coolant Accident

   This amendment request includes changes to the Technical Specifications based on assumptions in the accident analyses. The results of these analyses demonstrate that, with the requested changes, the dose consequences of these limiting events are within the regulatory limits provided by the NRC for use with the alternative source term.

   A new license and design basis analysis on secondary containment drawdown is provided to resolve a Justification for Continued Operation. The consequences, based on alternative source term methodology, remain within regulatory limits. This change to the licensing and design basis does not result in a significant increase in consequences.

Alternative source term methodology has been applied to resolve the Unresolved Safety Question on control room unfiltered air inleakage. The accident analyses results show, with the increased unfiltered air inleakage, the control room operator doses remain within regulatory limits.

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

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

   Response: No.

   The requested changes are based on accident analyses. System structure and component performance assumptions included in the accident analyses result in doses within regulatory limits. Use of these performance assumptions does not:

   - require the installation of any new equipment,
   - require the modification of any existing equipment,
   - change the manner in which the equipment is required to be operated,
   - assume equipment performance outside existing design capabilities, or
   - require new operator actions.

   Therefore Energy Northwest application of the alternative source term methodology does not create any new accident initiators or precursors of a new or different kind of accident.

3. Does the proposed amendment involve a significant reduction in the margin of safety?

   Response: No.

   The changes proposed are associated with the implementation of a new licensing basis for Columbia Generating Station. Approval of a basis change from the original source term developed in accordance with TID-14844 to a new alternative source term as described in RG 1.183 is requested. The results of the accident analyses revised in support of this submittal, and the requested Technical Specification changes, are subject to revised acceptance criteria. These analyses have been performed using conservative methodologies.

Safety margins and analytical conservatisms have been evaluated and are satisfied. The analyzed accidents have been carefully selected and margin has been retained to ensure that the analyses adequately bound postulated event scenarios. The dose consequences of these limiting design basis accidents are within the acceptance criteria found in the applicable regulatory requirements and guidance. These requirements and guidance are presented in 10 CFR 50, App. A, 10 CFR 50.67, GDC 19, and RG 1.183.

The proposed changes can be made while still satisfying regulatory requirements and review criteria, with margin. The changes continue to ensure that the doses at the exclusion area and low population zone boundaries, as well as the control room, are within the corresponding regulatory limits. Therefore, operation of Columbia Generating Station in accordance with the requested amendment does not involve a significant reduction in the margin of safety.

In summary and based upon the above considerations, Energy Northwest has concluded that a significant hazard would not be introduced as a result of this proposed change.

6.0 ENVIRONMENTAL CONSIDERATIONS

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21.

Energy Northwest has determined that the proposed change meets the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). The requested change does not involve a significant hazards consideration and does not involve a significant change in the types or significant increase in the amounts of any effluents that may be released off-site. The following table compares accident analysis dose results to the regulatory limits of 10 CFR 50.67 for the exclusion area boundary (EAB), the low population zone (LPZ) and control room. The calculated EAB and LPZ doses are a small fraction of the dose limits. The calculated control room operator doses are less than the TEDE limit (5 rem) over 30 days for all accidents.

DOSE RESULTS (rem)			
Accident	CR	EAB	LPZ
Regulatory Limit (TEDE)	5.0 rem	25 rem	25 rem
LOCA	3.5 rem	4.1 rem	4.0 rem
MSLB (pre-accident iodine spiking case)	1.8 rem	0.40 rem	0.11 rem
Regulatory Limit (TEDE)	5.0 rem	6.3 rem	6.3 rem
CRDA	0.7 rem	0.03 rem	0.03 rem
FHA	3.7 rem	1.0 rem	0.3 rem

Adoption of the alternative source term and Technical Specification changes, which implement certain conservative assumptions, do not result in modifications to the plant or changes in its operation which could alter the type or amounts of effluents that may be released offsite.

The alternative source term does not affect the design or operation of the facility; rather, once the occurrence of an accident has been postulated, the alternative source term is an input to evaluate the consequences of accidents. The implementation of the alternative source term has been evaluated in revisions to the analyses of the limiting design basis accidents at Columbia Generating Station (control rod drop accident, fuel handling accident, loss of coolant accident, and main steam line break accident). Based upon the results of these analyses it has been demonstrated that, with the requested changes, the dose consequences are within NRC regulatory limits for alternative source term (i.e., 10 CFR 50.67 and 10 CFR 50, Appendix A, General Design Criterion 19). Therefore, there is no significant increase in either individual or cumulative occupational radiation exposure.

7.0 REFERENCES

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

2. NUREG-0800, Section SRP 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

3. NUREG-1434, "Standard Technical Specifications General Electric Plants, BWR/6," Revision 3, June 2004

4. Generation Of Thermal Hydraulic Information for Containment (GOTHIC) Containment Analysis Package, Numerical Applications, Inc., Version 7.1

5. NUREG-0800, Section SRP 6.2.3, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000

6. Letter dated February 12, 2003, AS Bhatnagar (Tennessee Valley Authority), to NRC, "Browns Ferry Nuclear Power Plant (BFN) – Units 1, 2, and 3 – Technical Specifications (TS) Change 405 Supplement 1 – Decay Time (TAC Nos. MB5733, MB5734, MB5735)"

7. Letter dated September 29, 1989, GC Sorensen (Washington Public Power Supply System) to NRC, "Unreviewed Safety Question Regarding Standby Gas Treatment"

8. Letter dated December 6, 2000, RL Webring (Energy Northwest), to NRC, "Licensee Event Report No. 2000-006-01"

9. NRC Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," May 2003

10. Letter dated December 30, 1988, CM Powers (Washington Public Power Supply System) to NRC, "Licensee Event Report No. 88-023-01"

11. Letter dated January 3, 1990, RB Samworth (NRC) to GC Sorensen (Washington Public Power Supply System), Evaluation of JCO Regarding Standby Gas Treatment System Attainment of Secondary Containment Pressure (TAC No. 75048)"

12. Letter dated February 16, 1990, GC Sorensen (Washington Public Power Supply System) to NRC, "Standby Gas Treatment System (TAC No. 75048)"

13. Letter dated December 22, 1992, GC Sorensen (Washington Public Power Supply System) to NRC, "Standby Gas Treatment/Secondary Containment (TAC No. M75048)"

14. Letter dated October 15, 1996, PR Bemis (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

15. Letter dated December 4, 1997, DW Coleman (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications (Additional Information)"

16. Letter dated April 12, 1999, DW Coleman (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications (Additional Information)"

17. Letter dated June 10, 1999, DW Coleman (Washington Public Power Supply System) to NRC, "Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications (Additional Information)"

18. Letter dated July 16, 1999, RL Webring (Washington Public Power Supply System) to NRC, "Withdrawal of Request for Amendment to Secondary Containment and Standby Gas Treatment System Technical Specifications"

19. Letter dated December 3, 2001, RL Webring (Energy Northwest) to NRC, "License Amendment Request – Alternative Source Term"

20. Letter dated November 20, 2002, DK Atkinson (Energy Northwest) to NRC, "Withdrawal of Alternative Source Term License Amendment Request"

21. Letter GO2-01-156, dated December 3, 2001, RL Webring (Energy Northwest) to NRC, "License Amendment Request – Alternative Source Term"

22. ASTM E741-2000, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution"

23. NUREG-6331, "Atmospheric Relative Concentrations in Building Wakes," Rev. 1, May 1997, ARCON96, RSICC Computer Code Collection No. CCC-664

24. NUREG-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Bases Accidental Releases of Radioactive Materials from Nuclear Power Stations," November 1982, RSICC Computer Code Collection No.CCC-445

25. NRC Safety Guide 23, "Onsite Meteorological Programs" dated February 17, 1972

26. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments of Nuclear Power Plants," Revision 2

27. NRC Standard Review Plan Section 15.6.5, "Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Main Steam Isolation Valve Leakage Control System (BWR)"

28. AEB-98-03, "Assessment of Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term," December 9, 1998

29. "STARDOSE Model Report," Polestar Applied Technology, Inc., PSATCI09.03, dated January 1997

30. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation," April 1998 and Supplement 1, dated June 8, 1999

31. Energy Northwest Calculation, NE-02-01-13 Appendix B, "CR Shine Analysis Results - Control Room Shine Dose Calculation," Revision 0, dated October 31, 2001

32. Letter dated September 12, 2002, GS Vissing, Sr. (NRC) to M Kansler (Entergy Nuclear Operations), "James A. Fitzpatrick Nuclear Power Plant – Amendment Re: Technical Specification Change to the Requirements for Handling Irradiated Fuel Assemblies (TAC NO. MB5328)"

33. NRC issued Review Guidelines, "Guidance on the Assessment of a BWR SLC System for pH Control", dated February 12, 2004

34. Polestar Applied Technology, Inc., "STARpH Code Description and Validation and Verification Report," Document No. PSAT C107.02, Revision 4, dated February 16, 2000

Regulatory Guide 1.183 Comparison Matrix

Table 1. Comparison with Regulatory Guide 1.183 Main Sections			
RG Sec	RG Position	Columbia Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 17) or ORIGEN-ARP (Ref. 18). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2-based. Long-lived isotopes adjusted for 24-month cycle. Power level used = 3556 MW(t) to account for two percent uncertainty (3486 x 1.02 = 3556).
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking Factors of 1.7 are used for DBA events that do not involve the entire core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.	Conforms	A decay time of 24 hours is assumed for the FHA analysis.

<table>
<tr><th colspan="4">Table 1. Comparison with Regulatory Guide 1.183 Main Sections</th></tr>
<tr><th>RG Sec</th><th>RG Position</th><th>Columbia Analysis</th><th>Comments</th></tr>
<tr><td>3.2</td><td>The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

Table 1
BWR Core Inventory Fraction
Released Into Containment
<table><tr><th>Group</th><th>Gap Release Phase</th><th>Early In-Vessel Phase</th><th>Total</th></tr><tr><td>Noble Gases</td><td>0.05</td><td>0.95</td><td>1.0</td></tr><tr><td>Halogens</td><td>0.05</td><td>0.25</td><td>0.3</td></tr><tr><td>Alkali Metals</td><td>0.05</td><td>0.20</td><td>0.25</td></tr><tr><td>Tellurium Metals</td><td>0.00</td><td>0.05</td><td>0.05</td></tr><tr><td>Ba, Sr</td><td>0.00</td><td>0.02</td><td>0.02</td></tr><tr><td>Noble Metals</td><td>0.00</td><td>0.0025</td><td>0.0025</td></tr><tr><td>Cerium Group</td><td>0.00</td><td>0.0005</td><td>0.0005</td></tr><tr><td>Lanthanides</td><td>0.00</td><td>0.0002</td><td>0.0002</td></tr></table></td><td>Conforms</td><td>The fractions from Table 1 are used.</td></tr>
<tr><td>3.2</td><td>For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor.

Table 3
Non-LOCA Fraction of Fission Product Inventory in Gap
<table><tr><th>Group</th><th>Fraction</th></tr><tr><td>I-131</td><td>0.08*</td></tr><tr><td>Kr-85</td><td>0.10</td></tr><tr><td>Other Noble Gases</td><td>0.05*</td></tr><tr><td>Other Halogens</td><td>0.05*</td></tr><tr><td>Alkali Metals</td><td>0.12</td></tr></table>
* Per footnote 11, the fractions for Iodines and Noble Gas for a BWR rod drop accident are assumed to be 0.10.</td><td>Conforms</td><td>Peaking Factors of 1.7 are used for DBA events that do not involve the entire core.

For accidents with fractions different from Table 3, the accident-specific instructions given in the accident-specific appendix were followed.</td></tr>
</table>

Table 1. Comparison with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the duration of the phase. **Table 4** **LOCA Release Phases** **BWRs** **Phase** / **Onset** / **Duration** Gap Release / 2 min / 0.5 hr Early In-Vessel / 0.5 hr / 1.5 hr	Conforms	The BWR durations from Table 4 are used. LOCA – modeled in a linear fashion.
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable to the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	Columbia is not licensed to use the leak-before-break methodology for DBA analysis.
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** / **Elements** Noble Gases / Xe, Kr Halogens / I, Br Alkali Metals / Cs, Rb Tellurium Group / Te, Sb, Se, Ba, Sr Noble Metals / Ru, Rh, Pd, Mo, Tc, Co Lanthanides / La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium / Ce, Pu, Np	Meets The Intent	Barium and strontium have release fractions lower than the Te group, (see Item 3.2), and these fractions are used in lieu of the five percent release for the Te group. The nuclides used for Columbia are the 60 identified as being potentially important contributors to TEDE in NUREG/CR-4691 (MACCS Users Guide) [less the two cobalt isotopes which have a minor impact] plus four additional noble gas isotopes from TID-14844, plus three other short-lived noble gas isotopes, plus Ba137m for a total of 66.

Table 1. Comparison with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.	Conforms	
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	Enthalpy deposition postulated for CRDA. Mechanical damage for FHA.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE calculated. Significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	

Table 1. Comparison with Regulatory Guide 1.183 Main Sections			
RG Sec	RG Position	Columbia Analysis	Comments
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	DCFs taken from Federal Guidance Report 11 and 12 as represented by the default FGR11&12 file in NUREG/CR-6604.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release (see also Table 6).	Conforms	
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	

Table 1. Comparison with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: • Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, • Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, • Radiation shine from the external radioactive plume released from the facility, • Radiation shine from radioactive material in the reactor containment, • Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	First two items included in combination of filtered make-up and conservative overstatement of measured unfiltered inleakage. Last three items shown to be negligible.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.	Conforms	The source term, transport, and release methodology is the same for both the control room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Pressurization and intake filtration are credited in LOCA analysis only.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.

Table 1. Comparison with Regulatory Guide 1.183 Main Sections			
RG Sec	RG Position	Columbia Analysis	Comments
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $$DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$$	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms with the exception that the TID-14844 source term will continue to be used as the basis for the equipment qualification program.	Offsite and control room dose consequences were calculated using the guidance provided in Regulatory Positions 4.1 and 4.2 respectively. Doses to personnel in the TSC and EOF were qualitatively assessed as bounded by existing license basis analyses. The AST submittal requires no new operator actions thus there is no new operator dose that is not reported in current analyses. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," will continue to be used as the radiation dose basis for equipment qualification and radiation zone maps/shielding calculations.

Table 1. Comparison with Regulatory Guide 1.183 Main Sections

RG Sec	RG Position	Columbia Analysis	Comments
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.	Conforms	Credited mitigation features meet these requirements and are automatic except RHR drywell sprays and SLC injection. These manual actions are/will be explicitly addressed in emergency operating procedures. Loss of offsite power is assumed to occur concurrently with the initiation of each analyzed event.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis.	Conforms	
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	
5.3	Atmospheric dispersion values (X/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining X/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Refs. 6, 7, 22, and 28). The methodology of the NRC computer code ARCON96 (Ref 26) is generally acceptable to the NRC staff for use in determining control room X/Q values.	Conforms	New dispersion values included in submittal. Determination consistent with RG 1.145 for offsite using PAVAN. ARCON96 was used to determine the control room values except for MSLB. The MSLB control room X/Q is calculated using the puff methodology of RG 1.194.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used. An evaluation has been done to demonstrate pH > 7.0.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.	Conforms	Flow from the drywell to the wetwell has been ignored prior to the assumed core quench at two hours. Ignoring this flow is conservative since by remaining in the drywell, it contributes to MSIV leakage. For several minutes after the core quench, flow from the drywell to the wetwell would be expected to occur, and approximately half of the drywell contents would be expected to be purged into the wetwell at this time. Beyond the end of this purge flow, the drywell and wetwell gas spaces are assumed to be well-mixed.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	No credit taken for natural deposition.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3).	Conforms	SRP 6.5.2 model used. Elemental iodine assumed to be removed at the same rate as particulate. A study was also completed using an elemental iodine removal lambda of 20 per hour and a maximum elemental iodine DF of 121 (that corresponding to a pH of >7.3 not reached until 30 days which is conservative relative to the maximum DF of 200 permitted independent of pH). That study showed that the control room dose becomes about one percent lower than the value reported in the Columbia AST LAR. Therefore, the approach of treating elemental iodine as particulate is a conservative representation of the situation in which some elemental iodine would be removed by diffusion to spray water droplets (experiencing a removal lambda of 20 per hour, but a DF limit of between 121 and 200) and some elemental iodine would adsorb onto particulate (experiencing a lambda of 6.2 per hour until 98% of the particulate is removed, 0.62 per hour thereafter, and no maximum DF). Given this understanding, the method results in a slightly conservative dose and meets the intent of the regulatory position in Appendix A, Section 3.3, of RG 1.183 with comparable results.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
3.3	The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.	Conforms	Drywell assumed to be well-mixed based on the fact that the drywell is sufficiently small and the spray flowrate is sufficiently large. The ratio of spray flow to volume sprayed is 20-40 times larger for the Columbia drywell than for a typical sprayed region of a PWR. The mixing by momentum exchange alone (between the droplets and the atmosphere) will keep the drywell well-mixed; i.e., natural convection will play no noticeable role. Drywell congestion is explicitly addressed by reduced spray flow and fall height credit.
3.3	The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).	The conservative approach used in the Columbia analysis meets the intent of this guidance.	The SRP spray lambda is calculated per the SRP method. A reduction of 10 is taken when 98% of the particulate has been removed. Regarding the maximum DF for elemental iodine, please refer to the comment provided for Appendix A Section 3.3 two rows above.
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not applicable	The Columbia design does not include an in-containment recirculation filter system.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms	Pool scrubbing not credited.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not applicable	
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.	Conforms	Leakage reduced after 24 hours based upon reduced containment pressure. Primary containment pressure not brought subatmospheric.
3.7	For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.	Not applicable	Colombia has a Mark II type containment.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Not applicable	The primary containment is not routinely purged during power operation.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.	Conforms	No elevated release point at Columbia and no credit taken for elevated release.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	Assumed ground-level release directly to environment, unfiltered until secondary containment reaches technical specification pressure. Then secondary containment bypass leakage release and filtered release through standby gas treatment, both as ground level release.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	Certified 5%/95% meteorological data are used in analyzing the bounding conditions.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.	Conforms	No reactor building dilution credit is taken.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms	Bypass leakage rate based on the limit specified in the Technical Specifications.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.	Conforms	Fission products mixed into suppression pool during release.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	ESF leakage is assumed to begin at the time drywell sprays are started. TS 5.5.2 requires a leakage control program. 1 gpm is the licensing basis value at Columbia. The analysis assumes 2 gpm. Liquid leakage to the CST has been evaluated and shown to have a negligible contribution to dose.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ Where: h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.	Not applicable	The leakage temperature does not exceed 212°F.
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	A release fraction of 10% is assumed.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Credit is not taken for holdup and dilution of ESF leakage in reactor building and for release through SGT SYSTEM filters. Filter systems comply with RG 1.52 and GL 99-02.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	Postulated leakage rate was reduced by 50% after 24 hours.

Table 2. Comparison with Regulatory Guide 1.183 Appendix A (LOCA)

App Sec	RG Position	Columbia Analysis	Comments
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	AEB-98-03 well-mixed model used to calculate main steam line aerosol deposition. Deposition velocity reduced to take into account aerosol removal by spray in the drywell. RADTRAD Bixler models used for gaseous iodine deposition but in modified form to reflect well-mixed assumption instead of plug-flow assumption.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	MSIV leakage unprocessed, ground level release.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.	Conforms	No credit taken for qualified steam lines beyond outboard MSIVs.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	No purge assumed.

Table 3. Comparison with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	.
1.1	The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.	Conforms	
1.2	The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, halogens, cesiums, and rubidiums.	Conforms	Cesium and rubidium not included because DF assumed to be infinite (see response to Section 3 below).
1.3	The chemical form of radioiodine released from the fuel to the spent fuel pool should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. The CsI released from the fuel is assumed to completely dissociate in the pool water. Because of the low pH of the pool water, the iodine re-evolves as elemental iodine. This is assumed to occur instantaneously. The NRC staff will consider, on a case-by-case basis, justifiable mechanistic treatment of the iodine release from the pool.	Conforms	All iodine added to pool assumed to dissociate.
2	If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.85%) and organic iodine (0.15%) species results in the iodine above the water being composed of 57% elemental and 43% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1).	Conforms	Overall DF of 200 applied to iodine. Speciation after decontamination is 57% elemental and 43% organic.
3	The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).	Conforms	
4.1	The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.	Conforms	

Table 3. Comparison with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
4.2	A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system(21) should be determined and accounted for in the radioactivity release analyses.	Not applicable	No credit is taken for filtration from reactor building (i.e., SGT system). Additionally, no credit is taken for the control room filter (i.e., CREF).
4.3	The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.	Not applicable	No ESF filtration system is credited in the Columbia analysis. The release is postulated to occur over a 2-hour period in accordance with RG position 5.3 below.
5.1	If the containment is isolated during fuel handling operations, no radiological consequences need to be analyzed.	Not applicable	Containment not isolated.
5.2	If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment, no radiological consequences need to be analyzed.	Not applicable	Containment not isolated.
5.3	If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open), the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.	Conforms	
5.4	A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.	Not applicable	No credit being taken for filtration of release from reactor building.

Table 3. Comparison with Regulatory Guide 1.183 Appendix B (FHA)

App Sec	RG Position	Columbia Analysis	Comments
5.5	Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.	Conforms	Two-hour release to the environment assumed.

Table 4. Comparison with Regulatory Guide 1.183 Appendix C (CRDA)

App Sec	RG Position	Columbia Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.	Conforms	100% of the noble gases and 50% of the iodines released from melted fuel. Other releases also based on Regulatory Position 3 of main report.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 µCi/gm DE I-131) allowed by the technical specifications.	Conforms	More than minimal fuel damage is postulated. Coolant activity neglected.
3.1	The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.	Conforms	
3.2	Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.	Conforms	

Table 4. Comparison with Regulatory Guide 1.183 Appendix C (CRDA)

App Sec	RG Position	Columbia Analysis	Comments
3.3	Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.	Conforms	
3.4	Of the activity that reaches the turbine and condenser, 100% of the noble gases, 10% of the iodine, and 1% of the particulate radionuclides are available for release to the environment. The turbine and condensers leak to the atmosphere as a ground- level release at a rate of 1% per day for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.	Conforms	Release rate of 1% per day for 24 hours. Decay assumed in condenser.
3.5	In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.	Not applicable	Assumptions in paragraphs 3.2 through 3.4 were used.
3.6	The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic. The release from the turbine and condenser should be assumed to be 97% elemental and 3% organic.	Conforms	Release to environment assumed to be 97% elemental, 3% organic.
Foot-Note 1	The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.	Conforms	Projected fuel damage is the limiting case.
Foot-Note 2	If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.	Conforms	Forced flow paths of mechanical vacuum pumps and steam jet air ejectors automatically isolate; therefore, the release rate of 1% per day is acceptable.

Table 5. Comparison with Regulatory Guide 1.183 Appendix D (MSLB)

App Sec	RG Position	Columbia Analysis	Comments
1	Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.	Conforms	No fuel damage, release estimate based on coolant activity.
2	If no or minimal fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.	Conforms	4 μCi/gm consistent with spiking Tech Spec.
2.1	The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and *corresponds to the conditions of an assumed pre-accident spike, and*	Conforms	
2.2	The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.	Conforms	
3	The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.	Conforms	
4.1	The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.	Conforms	The 6 sec assumed in analysis is longer than the Tech Spec max closing time of 5 sec.
4.2	The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.	Conforms	
4.3	All the radioactivity in the released coolant should be assumed to be released to the atmosphere instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.	Conforms	Instantaneous puff release in accordance with RG 1.194.
4.4	The iodine species released from the main steam line should be assumed to be 95% CsI as an aerosol, 4.85% elemental, and 0.15% organic.	Conforms	

Proposed Technical Specification Changes (marked up)

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3.9.10 Decay Time 3.9.10-1

1.1 Definitions (continued)

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarm, interlock, display, and trip functions, and channel failure trips. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated core cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The COLR is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications.

Total Effective Dose Equivalent (TEDE)

DOSE EQUIVALENT I-131

DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same ~~thyroid~~ dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The ~~thyroid~~ dose conversion factors used for this calculation shall be those listed in ~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for~~

Federal Guidance Report (FGR) 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1988.

(continued)

1.1 Definitions

~~DOSE EQUIVALENT I-131 (continued)~~	~~Power and Test Reactor Sites;" Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977; or ICRP 30, Supplement to Part 1, page 192-212, Table titled "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."~~
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial signal generation by the associated turbine throttle valve limit switch or from when the turbine governor valve hydraulic control oil pressure drops below the pressure switch setpoint to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, ~~and~~ 2, and 3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SLC subsystem inoperable.	A.1 Restore SLC subsystem to OPERABLE status.	7 days
B. Two SLC subsystems inoperable.	B.1 Restore one SLC subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.7.1 Verify available volume of sodium pentaborate solution is ≥ 4587 gallons.	24 hours

(continued)

Table 3.3.6.1-1 (page 3 of 4)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RWCU System Isolation (continued)					
b. Differential Flow – Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 46.5 seconds
c. Blowdown Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 271.7 gpm
d. Heat Exchanger Room Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 160°F
e. Heat Exchanger Room Area Ventilation Differential Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 70°F
f. Pump Room Area Temperature – High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
g. Pump Room Area Ventilation Differential Temperature – High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 100°F
h. RWCU/RCIC Line Routing Area Temperature – High	1,2,3	1	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 180°F
i. RWCU Line Routing Area Temperature – High	1,2,3	1 per room	F	SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.6	
Room 409, 509 Areas					≤ 175°F
Room 408, 511 Areas					≤ 180°F
j. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≥ -58 inches
k. SLC System Initiation	1,2,3	2(c)	I	SR 3.3.6.1.6	NA
l. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.6	NA

(continued)

(c) SLC System Initiation only inputs into one of the two trip systems.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES AND OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3,(a)	2(c)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≥ -58 inches
2. Drywell Pressure – High	1,2,3	2(c)	SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation – High	1,2,3, (a),(b)	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.3 SR 3.3.6.2.4	≤ 16.0 mR/hr
4. Manual Initiation	1,2,3, (a),(b)	4	SR 3.3.6.2.4	NA

(a) During operations with a potential for draining the reactor vessel.

(b) ~~During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.~~

(c) Also required to initiate the associated LOCA Time Delay Relay ~~Funciton~~ pursuant to LCO 3.3.5.1.

Deleted

Function

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	E.1	-----NOTE----------- Enter applicable Conditions and Required Actions of LCO 3.7.3, "Control Room Emergency Filtration (CREF) System," when both remote air intakes are isolated. --------------------	
		Isolate the associated remote air intake.	1 hour from discovery of loss of radiation monitoring capability in a remote air intake
	AND		
	E.2	Restore channel to OPERABLE status.	7 days from discovery of inoperable channels associated with both remote air intakes AND 30 days
F. Required Action and associated Completion Time of Condition E not met.	F.1	Declare both CREF subsystems inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREF System Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREF initiation or radiation monitoring capability as applicable.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

Table 3.3.7.1-1 (page 1 of 1)
Control Room Emergency Filtration System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level – Low Low, Level 2	1,2,3, (a)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≥ -58 inches
2. Drywell Pressure – High	1,2,3	2	C	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 1.88 psig
3. Reactor Building Vent Exhaust Plenum Radiation – High	1,2,3, (a),(b)	2	B	SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3 SR 3.3.7.1.4	≤ 16.0 mR/hr
~~4. Main Control Room Ventilation Radiation Monitor~~	~~1,2,3, (a),(b)~~	~~2 per intake~~	~~E~~	~~SR 3.3.7.1.1 SR 3.3.7.1.2 SR 3.3.7.1.3~~	~~≤ 3800 cpm~~

(a) During operations with a potential for draining the reactor vessel.

~~(b) During CORE ALTERATIONS, and during movement of irradiated fuel assemblies in the secondary containment.~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.6	Verify the isolation time of each MSIV is ≥ 3 seconds and ≤ 5 seconds.[1]	In accordance with the Inservice Testing Program
SR 3.6.1.3.7	Verify each automatic PCIV actuates to the isolation position on an actual or simulated isolation signal.	24 months
SR 3.6.1.3.8	Verify a representative sample of reactor instrument line EFCVs actuate to the isolation position on an actual or simulated instrument line break signal.	24 months
SR 3.6.1.3.9	Remove and test the explosive squib from each shear isolation valve of the TIP System.	24 months on a STAGGERED TEST BASIS
SR 3.6.1.3.10	Verify the combined leakage rate for all secondary containment bypass leakage paths is ≤ ~~0.74 scfh~~ 0.04% primary containment volume/day when pressurized to ≥ P_a.	In accordance with the Primary Containment Leakage Rate Testing Program

(continued)

~~[1]The isolation time of each MSIV includes circuit response time and valve motion time. In addition, the fastest isolation times (excluding circuit response times) of the four main steam lines, when averaged together, shall be ≥ 3 seconds. This modification of SR 3.6.1.3.6 is effective until startup from refueling outage R-16 or startup from a forced outage of sufficient duration (> 72 hours) to perform testing to comply with SR 3.6.1.3.6 whichever occurs first.~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.6.1.3.11 Verify leakage rate through each MSIV is ≤ ~~11.5~~ scfh when tested at ≥ 25.0 psig. 16.0	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.12 Verify combined leakage rate through hydrostatically tested lines that penetrate the primary containment is within limits.	In accordance with the Primary Containment Leakage Rate Testing Program

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Main Steam Isolation Valve Leakage Control (MSLC) System

LCO 3.6.1.8 Two MSLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One MSLC subsystem inoperable.	A.1	Restore MSLC subsystem to OPERABLE status.	30 days
B. Two MSLC subsystems inoperable.	B.1	Restore one MSLC subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.1	Operate each MSLC blower ≥ 15 minutes.	31 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.8.2	Verify electrical continuity of each inboard MSLC subsystem heater element circuitry.	31 days
SR 3.6.1.8.3	Perform a system functional test of each MSLC subsystem.	18 months

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1	-----NOTE----- LCO 3.0.3 is not applicable -------------	
		Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>		
	C.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	C.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1.1	Verify secondary containment vacuum is ~~≥ 0.25~~ > 0.0 inch of vacuum water gauge.	24 hours
SR 3.6.4.1.2	Verify all secondary containment equipment hatches are closed and sealed.	31 days
SR 3.6.4.1.3	Verify each secondary containment access inner door or each secondary containment access outer door in each access opening is closed.	31 days
~~SR 3.6.4.1.4~~	~~Verify each standby gas treatment (SGT) subsystem will draw down the secondary containment to ≥ 0.25 inch of vacuum water gauge in ≤ 120 seconds.~~	~~24 months on a STAGGERED TEST BASIS~~
SR 3.6.4.1.~~5~~ 4	Verify each SGT subsystem can maintain ≥ 0.25 inch of vacuum water gauge in the secondary containment for 1 hour at an inleakage flow rate ≤ ~~2240~~ 2430 cfm.	24 months on a STAGGERED TEST BASIS

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

------------------------------------NOTES------------------------------------

1. Penetration flow paths may be unisolated intermittently under administrative controls.

2. Separate Condition entry is allowed for each penetration flow path.

3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange. AND	8 hours
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met during ~~movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs.	D.1	~~-------NOTE---------~~ ~~LCO 3.0.3 is not applicable.~~ ~~---------------------~~ ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	<u>AND</u>		
	D.2	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	<u>AND</u>		
	D.3	Initiate action to suspend OPDRVs.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 Two SGT subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One SGT subsystem inoperable.	A.1 Restore SGT subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during ~~movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during~~ OPDRVs.	~~-----NOTE----- LCO 3.0.3 is not applicable.~~ C.1 Place OPERABLE SGT subsystem in operation. OR	Immediately
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. (continued)	C.2.1	Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	C.2.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	C.2.3	Initiate action to suspend OPDRVs.	Immediately
D. Two SGT subsystems inoperable in MODE 1, 2, or 3.	D.1	Enter LCO 3.0.3.	Immediately
E. Two SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	E.1	-----NOTE----- LCO 3.0.3 is not applicable. ------------ Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND		
	E.2	Suspend CORE ALTERATIONS.	Immediately
	AND		
	E.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each SGT subsystem for ≥ 10 continuous hours with heaters operating.	31 days
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.6.4.3.3	Verify each SGT subsystem actuates on an actual or simulated initiation signal.	24 months
SR 3.6.4.3.4	Verify each SGT filter cooling recirculation valve can be opened and the fan started.	24 months

and reaches ≥ 4800 cfm within 2 minutes

3.7 PLANT SYSTEMS

3.7.3 Control Room Emergency Filtration (CREF) System

LCO 3.7.3 Two CREF subsystems shall be OPERABLE.

-------------------------NOTE-------------------------
The control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREF subsystem inoperable.	A.1 Restore CREF subsystem to OPERABLE status.	7 days
B. Two CREF subsystems inoperable due to inoperable control room boundary in MODES 1, 2, and 3.	B.1 Restore control room boundary to OPERABLE status.	24 hours
C. Required Action and Associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------------	
	D.1 Place OPERABLE CREF subsystem in pressurization mode.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately
E. Two CREF subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two CREF subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------	
	F.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	F.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	F.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Operate each CREF subsystem for $\geq$ 10 continuous hours with the heaters operating.	31 days
SR 3.7.3.2	Perform required CREF filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

3.7 PLANT SYSTEMS

3.7.4 Control Room Air Conditioning (AC) System

LCO 3.7.4 Two control room AC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
~~During movement of irradiated fuel assemblies in the secondary containment,~~
~~During CORE ALTERATIONS,~~
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One control room AC subsystem inoperable.	A.1	Restore control room AC subsystem to OPERABLE status.	30 days
B. Required Action and Associated Completion Time of Condition A not met in MODE 1, 2, or 3.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------NOTE----------- LCO 3.0.3 is not applicable. ------------------------------	
	C.1 Place OPERABLE control room AC subsystem in operation.	Immediately
	OR	
	C.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	C.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	C.2.3 Initiate action to suspend OPDRVs.	Immediately
D. Two control room AC subsystems inoperable in MODE 1, 2, or 3.	D.1 Enter LCO 3.0.3.	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two control room AC subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----NOTE----- LCO 3.0.3 is not applicable. ----------	
	E.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	E.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	E.3 Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each control room AC subsystem has the capability to remove the assumed heat load.	24 months

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources – Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems – Shutdown";

b. One diesel generator (DG) capable of supplying one division of the Division 1 or 2 onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.8; and

c. The Division 3 DG capable of supplying the Division 3 onsite Class 1E AC electrical power distribution subsystem, when the Division 3 onsite Class 1E electrical power distribution subsystem is required by LCO 3.8.8.

APPLICABILITY: MODES 4 and 5,
During movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

~~-----NOTE-----~~
~~LCO 3.0.3 is not applicable.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required offsite circuit inoperable.	-----NOTE----- Enter applicable Condition and Required Actions of LCO 3.8.8, when any required division is de-energized as a result of Condition A.	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	OR	
	A.2.1 ~~Suspend CORE ALTERATIONS.~~	~~Immediately~~
	AND	
	A.2.2 ~~Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	AND	
	A.2.3 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2.~~1~~ 2	Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. Division 1 or 2 required DG inoperable.	~~B.1~~	~~Suspend CORE ALTERATIONS.~~	~~Immediately~~ D
	~~AND~~		
	~~B.2~~	~~Suspend movement of irradiated fuel assemblies in secondary containment.~~	~~Immediately~~
	AND		
	B.~~3~~ 1	Initiate action to suspend OPDRVs.	Immediately
	AND		
	B.~~4~~ 2	Initiate action to restore required DG to OPERABLE status.	Immediately
C. Required Division 3 DG inoperable.	C.1	Declare High Pressure Core Spray System inoperable.	72 hours

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources – Shutdown

LCO 3.8.5 DC electrical power subsystem(s) shall be OPERABLE to support the electrical power distribution subsystem(s) required by LCO 3.8.8, "Distribution Systems – Shutdown."

APPLICABILITY: MODES 4 and 5,
~~During movement of irradiated fuel assemblies in the secondary containment.~~

ACTIONS

~~NOTE~~
~~LCO 3.0.3 is not applicable.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required DC electrical power subsystems inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	~~A.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~AND~~	
	~~A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~AND~~	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.~~2~~1 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.~~1~~2 Initiate action to restore required DC electrical power subsystems to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.5.1	-------------------NOTE------------------ The following SRs are not required to be performed: SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8. --- For DC electrical power subsystems required to be OPERABLE the following SRs are applicable: SR 3.8.4.1, SR 3.8.4.2, SR 3.8.4.3, SR 3.8.4.4, SR 3.8.4.5, SR 3.8.4.6, SR 3.8.4.7, and SR 3.8.4.8.	In accordance with applicable SRs

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Distribution Systems – Shutdown

LCO 3.8.8 The necessary portions of the Division 1, Division 2, and Division 3 AC and DC electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 4 and 5,
~~During movement of irradiated fuel assemblies in the secondary containment.~~

ACTIONS

-----------------------------------NOTE-----------------------------------

~~LCO 3.0.3 is not applicable.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC or DC electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	~~A.2.1 Suspend CORE ALTERATIONS.~~	~~Immediately~~
	~~<u>AND</u>~~	
	~~A.2.2 Suspend movement of irradiated fuel assemblies in the secondary containment.~~	~~Immediately~~
	~~<u>AND</u>~~	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.~~3~~ 1 Initiate action to suspend operations with a potential for draining the reactor vessel.	Immediately
	AND	
	A.2.~~4~~ 2 Initiate actions to restore required AC and DC electrical power distribution subsystems to OPERABLE status.	Immediately
	AND	
	A.2.~~5~~ 3 Declare associated required shutdown cooling subsystem(s) inoperable and not in operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	Verify correct breaker alignments and indicated power availability to required AC and DC electrical power distribution subsystems.	7 days

3.9 REFUELING OPERATIONS

3.9.7 Reactor Pressure Vessel (RPV) Water Level — New Fuel or Control Rods

LCO 3.9.7 RPV water level shall be ≥ ~~22~~ 23 ft above the top of irradiated fuel assemblies seated within the RPV.

APPLICABILITY: During movement of new fuel assemblies or handling of control rods within the RPV when irradiated fuel assemblies are seated within the RPV.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RPV water level not within limit.	A.1 Suspend movement of new fuel assemblies and handling of control rods within the RPV.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.9.7.1	Verify RPV water level is ≥ ~~22~~ 23 ft above the top of irradiated fuel assemblies seated within the RPV.	24 hours

New

3.9 REFUELING OPERATIONS

3.9.10 Decay Time

LCO 3.9.10 The reactor shall be subcritical for at least 24 hours.

APPLICABILITY: During in-vessel fuel movement.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. With the reactor subcritical for less than 24 hours.	A.1 Suspend in-vessel fuel movement.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.9.10.1 Verify the reactor has been subcritical for at least 24 hours.	Once prior to the movement of irradiated fuel in the reactor vessel.

4320 to 5280

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

ESF Ventilation System	Flowrate (cfm)
SGT System	~~4012 to 4902~~
CREF System	900 to 1100

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2, and ASME N510-1989 at the system flowrate specified below:

ESF Ventilation System	Flowrate (cfm)
SGT System	~~4012 to 4902~~
CREF System	900 to 1100

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C (86°F) and the relative humidity specified below. Testing of the SGT System will also be conducted at a face velocity of 75 feet per minute.

ESF Ventilation System	Penetration (%)	RH (%)
SGT System	0.5	70
CREF System	2.5	70

Allowed tolerances in the above testing parameters of temperature, relative humidity, and face velocity are as specified in ASTM D3803-1989.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below:

ESF Ventilation System	Delta P (inches wg)	Flowrate (cfm)
SGT System	< 8	~~4012 to 4902~~
CREF System	< 6	900 to 1100

(continued)

Proposed Technical Specification Base Changes (marked up)

B ii
B iii
B 2.1.1-5
B 2.1.2-1
B 2.1.2-2
B 2.1.2-3
B 3.1.6-5
B 3.1.7-1
B 3.1.7-2
B 3.1.7-3
B 3.1.7-6
B 3.1.8-1
B 3.1.8-5
B 3.2.3-1
B 3.3.6.1-8
B 3.3.6.1-9
B 3.3.6.1-12
B 3.3.6.1-14
B 3.3.6.1-16
B 3.3.6.1-25
B 3.3.6.2-1
B 3.3.6.2-2
B 3.3.6.2-5
B 3.3.6.2-6
B 3.3.6.2-7
B 3.3.6.2-8
B 3.3.6.2-10
B 3.3.6.2-12
B 3.3.7.1-1
B 3.3.7.1-2
B 3.3.7.1-4
B 3.3.7.1-5
B 3.3.7.1-6
B 3.3.7.1-9
B 3.3.7.1-10
B 3.3.7.1-11
B 3.4.8-1
B 3.4.8-2
B 3.4.8-3
B 3.4.8-4
B 3.6.1.1-2
B 3.6.1.1-5
B 3.6.1.3-2
B 3.6.1.3-15
B 3.6.1.5-1
B 3.6.1.5-4
B 3.6.1.8-1
B 3.6.1.8-2
B 3.6.1.8-3
B 3.6.1.8-4
B 3.6.4.1-1
B 3.6.4.1-2
B 3.6.4.1-3
B 3.6.4.1-4
B 3.6.4.1-5
B 3.6.4.1-6
B 3.6.4.2-1
B 3.6.4.2-2
B 3.6.4.2-3
B 3.6.4.2-5
B 3.6.4.2-6
B 3.6.4.2-7
B 3.6.4.2-8
B 3.6.4.3-1
B 3.6.4.3-2
B 3.6.4.3-3
B 3.6.4.3-4
B 3.6.4.3-5
B 3.6.4.3-6
B 3.6.4.3-7
B 3.7.3-1
B 3.7.3-2
B 3.7.3-3
B 3.7.3-5
B 3.7.3-6
B 3.7.3-8
B 3.7.4-2
B 3.7.4-3
B 3.7.4-4
B 3.7.4-5
B 3.7.4-6
B 3.7.5-1
B 3.7.5-3
B 3.7.7-1
B 3.7.7-3
B 3.8.2-1
B 3.8.2-3
B 3.8.2-4
B 3.8.2-5
B 3.8.2-6
B 3.8.5-1
B 3.8.5-2
B 3.8.5-3
B 3.8.5-4
B 3.8.8-1
B 3.8.8-2
B 3.8.8-3
B 3.8.8-4
B 3.8.8-5
B 3.9.6-1
B 3.9.6-2
B 3.9.6-3
B 3.9.7-1
B 3.9.7-2
B 3.9.7-3
B 3.9.10-1
B 3.9.10-2

(continued)

(continued)

B 3.9.10 Decay Time .... B 3.9.10-1

BASES

SAFETY LIMITS (continued)

SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR ~~100, "Reactor Site Criteria,"~~ 50.67 limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. EMF-2209(P)(A) Revision 1, "SPCB Critical Power Correlation," Siemens Power Corporation, July 2000.

3. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlation to Co-resident Fuel," Siemens Power Corporation, August 2000.

4. NE-02-02-15 Revision 0, "Computation of SPCB Critical Power Correlation Additive Constants for SVEA-96," November 2002.

5. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, "ANF Critical Power Methodology for Boiling Water Reactors," Advanced Nuclear Fuels, November 1990.

6. 10 CFR ~~100~~. 50.67, "Accident Source Term."

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on reactor steam dome pressure protects the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. Establishing an upper limit on reactor steam dome pressure ensures continued RCS integrity. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to ensure that the design conditions are not exceeded during normal operation and anticipated operational occurrences (AOOs).

During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, in accordance with ASME Code requirements, prior to initial operation when there is no fuel in the core. Any further hydrostatic testing with fuel in the core may be done under LCO 3.10.1, "Inservice Leak and Hydrostatic Testing Operation." Following inception of unit operation, RCS components shall be pressure tested in accordance with the requirements of ASME Code, Section XI (Ref. 3).

50.67

Overpressurization of the RCS could result in a breach of the RCPB reducing the number of protective barriers designed to prevent radioactive releases from exceeding the limits specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4). If this occurred in conjunction with a fuel cladding failure, the number of protective barriers designed to prevent radioactive releases from exceeding the limits would be reduced.

APPLICABLE SAFETY ANALYSES

The RCS safety/relief valves and the Reactor Protection System Reactor Vessel Steam Dome Pressure—High Function have settings established to ensure that the RCS pressure SL will not be exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is designed to ASME Code, Section III, 1971 Edition, including Addenda through the summer of 1971 (Ref. 5), for the reactor recirculation piping, which permits a maximum pressure transient of 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 125% of design pressures of 1250 psig for suction piping and 1550 psig for discharge piping. The most limiting of these allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured at the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

50.67

Exceeding the RCS pressure SL may cause RCS failure and create a potential for radioactive releases in excess of 10 CFR ~~100, Reactor Site Criteria,~~ limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.

3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

4. 10 CFR 100. 50.67, "Accident Source Term."

5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, summer of 1971.

BASES

REFERENCES (continued)

4. NUREG-0800, "Standard Review Plan," Section 15.4.9, "Radiological Consequences of Control Rod Drop Accident (BWR)," Revision 2, July 1981.

5. 10 CFR ~~100.11, "Determination of Exclusion Area Low Population Zone and Population Center Distance."~~

50.67, "Accident Source Term."

6. NEDO-10527, "Rod Drop Accident Analysis for Large BWRs," (including Supplements 1 and 2), March 1972.

7. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

8. ASME, Boiler and Pressure Vessel Code, Section III.

9. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

10. 10 CFR 50.36(c)(2)(ii).

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water (Ref. 4).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using both SLC pumps, a quantity of boron that produces a concentration of 660 ppm of natural boron in the reactor core, including recirculation loops, at 70°F and normal reactor water level. To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). An additional 275 ppm is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping. The temperature versus concentration limits in Figure 3.1.7-1

(continued)

Following a LOCA, offsite doses from the accident will remain within 10 CFR 50.67, "Accident Source Term," limits (Ref. 5) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Ref. 4) as long as suppression pool pH is maintained at or above 7. Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool at or above 7.

BASES

APPLICABLE SAFETY ANALYSES (continued)

are calculated such that the required concentration is achieved. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The SLC System satisfies Criterion 4 of ~~Reference~~ 3.

a 3 and

LCO

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

Additionally, an OPERABLE SLC system has the ability to inject boron under post LOCA conditions to maintain the suppression pool pH above 7.

~~APPLICABILITY~~

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to be OPERABLE during these conditions, when only a single control rod can be withdrawn.

perform its ATWS function during MODES 3, 4, or 5.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 5) limits following a LOCA involving significant fission product releases. The SLC System is used to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 4).

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the original licensing basis shutdown function. However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable

(continued)

BASES

ACTIONS

A.1 (continued)

of performing the original licensing basis SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

at least

C.1 and C.2

and to MODE 4 within 36 hours

If any Required Action and associated Completion Time is not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion ~~Time of 12 hours is~~ reasonable, based on operating experience, to reach ~~MODE 3~~ from full power conditions in an orderly manner and without challenging plant systems.

Times are

the required plant conditions

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 and SR 3.1.7.2

SR 3.1.7.1 and SR 3.1.7.2 are 24 hour Surveillances, verifying certain characteristics of the SLC System (e.g., the volume and temperature of the borated solution in the storage tank), thereby ensuring the SLC System OPERABILITY without disturbing normal plant operation. These Surveillances ensure the proper borated solution and temperature are maintained. Maintaining a minimum specified borated solution temperature is important in ensuring that the boron remains in solution and does not precipitate out in the storage tank. The 24 hour Frequency of these SRs is based on operating experience that has shown there are relatively slow variations in the measured parameters of volume and temperature.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 and SR 3.1.7.8 (continued)

potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance test when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction valve to the injection pumps is unblocked ensures that there is a functioning flow path for injecting the sodium pentaborate solution. An acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be drained and flushed with demineralized water since the suction piping between the pump suction valve and pump suction is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. However, if, in performing SR 3.1.7.1, it is determined that the temperature of the solution in the storage tank has fallen below the specified minimum, SR 3.1.7.8 must be performed once within 24 hours after the solution temperature is restored within the limits of Figure 3.1.7-1.

REFERENCES

1. 10 CFR 50.62.
2. FSAR, Section 9.3.5.3.
3. 10 CFR 50.36(c)(2)(ii).
4. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," USNRC, Febaruary 1995.
5. 10 CFR 50.67, "Accident Source Term."

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves

BASES

BACKGROUND

The SDV vent and drain valves are normally open and discharge any accumulated water in the SDV to ensure that sufficient volume is available at all times to allow a complete scram. During a scram, the SDV vent and drain valves close to contain reactor water. The SDV consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. There are two headers and two instrument volumes, each receiving approximately one half of the control rod drive (CRD) discharges. The two instrument volumes are connected to a common drain line with two valves in series. Each header is connected to a common vent line with two valves in series. The header piping is sized to receive and contain all the water discharged by the CRDs during a scram. The design and functions of the SDV are described in Reference 1.

APPLICABLE SAFETY ANALYSES

The Design Basis Accident and transient analyses assume all the control rods are capable of scramming. The primary function of the SDV is to limit the amount of reactor coolant discharged during a scram. The acceptance criteria for the SDV vent and drain valves are that they operate automatically to:

a. Close during scram to limit the amount of reactor coolant discharged so that adequate core cooling is maintained and offsite doses remain within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2); and

b. Open on scram reset to maintain the SDV vent and drain path open so there is sufficient volume to accept the reactor coolant discharged during a scram.

Isolation of the SDV can also be accomplished by manual closure of the SDV valves. Additionally, the discharge of reactor coolant to the SDV can be terminated by scram reset or closure of the HCU manual isolation valves. For a bounding leakage case, the offsite doses are well within the limits of 10 CFR ~~100~~ 50.67 (Ref. 2) and adequate core cooling is maintained (Ref. 3). The SDV vent and drain valves also allow continuous drainage of the SDV during normal plant

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.1.8.3 (continued) unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency; therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section 4.6.1.1.2.4.2.5. 2. 10 CFR ~~100~~ 50.67, "Accident Source Term." 3. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," August 1981. 4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 3, 4, 5, and 6. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR,

50.67

Parts 20, 50, and ~~100~~. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Reference 7).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Reactor Vessel Water Level – Low Low, Level 2 (continued)

recirculation line break (Ref. 1). The isolation of the MSL on Level 2 supports actions to ensure that offsite dose limits are not exceeded for a DBA.

Reactor vessel water level signals are initiated from four differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level – Low Low, Level 2 Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Reactor Vessel Water Level – Low Low, Level 2 Allowable Value is chosen to be the same as the ECCS Level 2 Allowable Value (LCO 3.3.5.1) to ensure that the MSLs isolate on a potential loss of coolant accident (LOCA) to prevent offsite doses from exceeding 10 CFR ~~100~~ limits. 50.67

This Function isolates the Group 1 valves.

1.b. Main Steam Line Pressure – Low

Low MSL pressure indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure – Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 4). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

The MSL low pressure signals are initiated from four sensors that are connected to the MSL header. The sensors are arranged such that, even though physically separated from each other, each sensor is able to detect low MSL pressure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure – Low (continued)

Four channels of Main Steam Line Pressure – Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure – Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 4).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow – High

Main Steam Line Flow – High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow – High Function is directly assumed in the analysis of the main stream line break (MSLB) accident (Ref. 5). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

50.67

The MSL flow signals are initiated from 16 differential pressure switches that are connected to the four MSLs (the differential pressure switches sense d/p across a flow restrictor). The differential pressure switches are arranged such that, even though physically separated from each other, all four connected to one steam line would be able to detect the high flow. Four channels of Main Steam Line Flow – High Function for each MSL (two channels per trip system) are available and are required to be OPERABLE so that no single instrument failure will preclude detecting a break in any individual MSL.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

<u>1.g. Manual Initiation</u> (continued)

It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

There are four switch and push buttons (with two channels per switch and push button) for the logic, with two switch and push buttons per trip system. Eight channels of Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the MSL Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

This Function isolates the Group 1 valves.

<u>2. Primary Containment Isolation</u>

<u>2.a, 2.b. Reactor Vessel Water Level — Low, Level 3 and Reactor Vessel Water Level — Low Low, Level 2</u>

Low RPV water level indicates the capability to cool the fuel may be threatened. The valves whose penetrations communicate with the primary containment are isolated to limit the release of fission products. The isolation of the primary containment on Level 3 and 2 supports actions to ensure that offsite dose limits of 10 CFR 100 are not 50.67 exceeded. The Reactor Vessel Water Level — Low, Level 3 and Reactor Vessel Water Level — Low Low, Level 2 Functions associated with isolation are implicitly assumed in the FSAR analysis as these leakage paths are assumed to be isolated post LOCA.

Reactor Vessel Water Level — Low, Level 3 and Reactor Vessel Water Level — Low Low, Level 2 signals are initiated from differential pressure switches that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level — Low, Level 3 Function and four channels of Reactor Vessel Water Level — Low Low, Level 2

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>2.a, 2.b. Reactor Vessel Water Level—Low, Level 3 and Reactor Vessel Water Level—Low Low, Level 2</u> (continued)

does not already result in the channel being in a tripped condition). If the 230 kV offsite source is supplying the safety buses, the LOCA Time Delay Relays will start timing out immediately and will no longer sequence the delay after HPCS pump starts. If the 230 kV offsite source is not supplying safety buses, the LOCA Time Delay Relays will begin timing out upon transfer to the 230 kV source supply rather than initiating on a LOCA signal at the same time because the HPCS pump starts from different reactor Level 2 instruments. In either case, the LOCA Time Delay Relays may not be properly sequenced to delay start of the low pressure ECCS subsystems tied to when the HPCS pump starts.

<u>2.c. Drywell Pressure—High</u> 50.67

High drywell pressure can indicate a break in the RCPB inside the drywell. The isolation of some of the PCIVs on high drywell pressure supports actions to ensure that offsite dose limits of 10 CFR 100 are not exceeded. The Drywell Pressure—High Function associated with isolation of the primary containment is implicitly assumed in the FSAR accident analysis as these leakage paths are assumed to be isolated post LOCA.

High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell Pressure—High are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be the same as the RPS Drywell Pressure—High Allowable Value (LCO 3.3.1.1), since this may be indicative of a LOCA inside primary containment.

The above Function isolates the Group 2, 3, 4, and 5 valves.

The Drywell Pressure—High Function is also used to initiate the LOCA Time Delay Relays of LCO 3.3.5.1. These LOCA Time Delay Relays stagger ECCS pump loading when the ECCS power source is aligned to the 230 kV offsite circuit to assure ECCS loading, during pump starts, does not overload the offsite source transformer. This branching to LCO 3.3.5.1

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Reactor Building Vent Exhaust Plenum Radiation—High (continued)

The Reactor Building Vent Exhaust Plenum Radiation—High signals are initiated from radiation detectors that are located in the ventilation exhaust plenum. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Vent Exhaust Plenum Radiation—High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

50.67

The Allowable Values are chosen to ensure offsite doses remain below 10 CFR 100 limits.

This Function isolates the Group 3 valves.

2.e. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the primary containment isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific FSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

For the Group 3 valves, there are four switch and push buttons (with two channels per switch and push button) for the logic, with two switch and push buttons per trip system. For the Group 2, 4, and 5 valves, there are two switch and push buttons (with two channels per switch and push button) for the logic, one switch and push button per trip system. Eight channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3, since these are the MODES in which the Primary Containment Isolation automatic Functions are required to be OPERABLE.

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.j. Reactor Vessel Water Level - Low Low, Level 2 (continued)

This Function isolates the Group 7 valves.

4.k. SLC System Initiation

The isolation of the RWCU System is required when the SLC System has been initiated to prevent dilution and removal of the boron solution by the RWCU System (Ref. 8). SLC System initiation signals are initiated from the two SLC pump start signals.

Two channels (one from each pump) of SLC System Initiation Function are available and are required to be OPERABLE ~~only~~ in MODES 1 and 2, since these are the only MODES where the reactor can be critical, and these MODES are consistent with the Applicability for the SLC System (LCO 3.1.7).

Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC system is used to maintain suppression pool pH at or above 7 following a LOCA to ensure iodine will be retained in the suppression pool water.

Compliance with Reference 9 (Columbia Generating Station requires both SLC pumps be started to inject boron) ensures no single instrument failure can preclude the isolation function. As noted (footnote (c) to Table 3.3.6.1-1), this Function is only required to close the outboard Group 7 RWCU isolation valve since the signal only provides input into one of the two trip systems.

There is no Allowable Value associated with this Function since the channels are mechanically actuated based solely on the position of the SLC System initiation switch.

This Function isolates the Group 7 valves.

4.l. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the RWCU System isolation logic that are redundant to the automatic protective instrumentation and provide manual isolation capability. There is no specific FSAR safety analysis that takes credit for this Function. It is retained for overall redundancy and diversity of the isolation function as required by the NRC in the plant licensing basis.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.6.2 Secondary Containment Isolation Instrumentation

BASES

BACKGROUND

50.67

The secondary containment isolation instrumentation automatically initiates closure of appropriate secondary containment isolation valves (SCIVs) and starts the Standby Gas Treatment (SGT) System. The function of these systems, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Ref~~s~~. 1 ~~and 2~~), such that offsite radiation exposures are maintained within the requirements of 10 CFR ~~100~~ that are part of the NRC staff approved licensing basis. Secondary containment isolation and establishment of vacuum with the SGT System within the assumed time limits ensures that fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment are maintained within applicable limits.

The isolation instrumentation includes the sensors, relays, and switches that are necessary to cause initiation of secondary containment isolation. Most channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs a secondary containment isolation signal to the isolation logic. Functional diversity is provided by monitoring a wide range of independent parameters. The input parameters to the isolation logic are (a) reactor vessel water level, (b) drywell pressure, and (c) reactor building vent exhaust plenum radiation. Redundant sensor input signals from each parameter are provided for initiation of isolation parameters. In addition, manual initiation of the logic is provided.

Most Secondary Containment Isolation instrumentation Functions receive input from four channels. The output from these channels are arranged into two two-out-of-two logic trip systems. For the Manual Initiation Function, four channels are required to actuate a trip system (a four-out-of-four logic trip system). In addition to the isolation function, the SGT subsystems are initiated. Each trip system will start one fan in each SGT subsystem, but

(continued)

BASES

BACKGROUND (continued)

will only align one SGT subsystem filter train. Automatically isolated secondary containment penetrations are isolated by two isolation valves. Each trip system initiates isolation of one of the two valves on each penetration so that operation of either trip system isolates the penetrations.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The isolation signals generated by the secondary containment isolation instrumentation are implicitly assumed in the safety analyses of References 1 and 2 to initiate closure of valves and start the SGT System to limit offsite doses.

Refer to LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System," Applicable Safety Analyses Bases for more detail of the safety analyses.

2

The secondary containment isolation instrumentation satisfies Criterion 3 of Reference 3. Certain instrumentation Functions are retained for other reasons and are described below in the individual Functions discussion.

The OPERABILITY of the secondary containment isolation instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions. Each Function must have the required number of OPERABLE channels with their setpoints set within the specified Allowable Values, as shown in Table 3.3.6.2-1. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each Function specified in the Table. Nominal trip setpoints are specified in setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. Drywell Pressure – High (continued)

supports actions to ensure that any offsite releases are within the limits calculated in the safety analysis. However, the Drywell Pressure – High Function associated with isolation is not assumed in any FSAR accident or transient analysis. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis. High drywell pressure signals are initiated from pressure switches that sense the pressure in the drywell. Four channels of Drywell – High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was chosen to be the same as the RPS Drywell Pressure – High Function Allowable Value (LCO 3.3.1.1) since this is indicative of a loss of coolant accident.

The Drywell Pressure – High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists in the RCS; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. This Function is not required in MODES 4 and 5 because the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES.

3. Reactor Building Vent Exhaust Plenum Radiation – High

~~-----------------------------NOTE-----------------------------~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~--~~

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Reactor Building Vent Exhaust Plenum Radiation – High is detected, secondary containment

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Building Vent Exhaust Plenum Radiation — High (continued)

isolation and actuation of the SGT System are initiated to limit the release of fission products as assumed in the FSAR safety analyses (Ref. 2).

The Reactor Building Vent Exhaust Plenum Radiation — High signals are initiated from radiation detectors that are located in the ventilation exhaust plenum, which is the collection point of all reactor building and refueling floor air flow prior to its exhaust to atmosphere. The signal from each detector is input to an individual monitor whose trip outputs are assigned to an isolation channel. Four channels of Reactor Building Vent Exhaust Plenum Radiation — High Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value is chosen to promptly detect gross failure of the fuel cladding.

The Reactor Building Vent Plenum Exhaust Radiation — High Function is required to be OPERABLE in MODES 1, 2, and 3 where considerable energy exists; thus, there is a probability of pipe breaks resulting in significant releases of radioactive steam and gas. In MODES 4 and 5, the probability and consequences of these events are low due to the RCS pressure and temperature limitations of these MODES; thus, this Function is not required. In addition, the Function is required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment because the capability of detecting radiation releases due to fuel failures (due to fuel uncovery or dropped fuel assemblies) must be provided to ensure that offsite dose limits are not exceeded.

4. Manual Initiation

The Manual Initiation switch and push button channels introduce signals into the secondary containment isolation logic that are redundant to the automatic protective instrumentation channels, and provide manual isolation capability. There is no specific FSAR safety analysis that

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. Manual Initiation (continued)

takes credit for this Function. It is retained for the overall redundancy and diversity of the secondary containment isolation instrumentation as required by the NRC approved licensing basis.

There are four switch and push buttons (with two channels per switch and push button) for the logic, two switch and push buttons per trip system. Eight channels of the Manual Initiation Function are available and are required to be OPERABLE in MODES 1, 2, and 3 and during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment, since these are the MODES and other specified conditions in which the Secondary Containment Isolation automatic Functions are required to be OPERABLE. the

There is no Allowable Value for this Function since the channels are mechanically actuated based solely on the position of the switch and push buttons.

ACTIONS

A Note has been provided to modify the ACTIONS related to secondary containment isolation instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable secondary containment isolation instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable secondary containment isolation instrumentation channel.

A.1

Because of the diversity of sensors available to provide isolation signals and the redundancy of the isolation design, an allowable out of service time of 12 hours or

(continued)

BASES

ACTIONS

A.1 (continued)

24 hours, depending on the Function (12 hours for those Functions that have channel components common to RPS instrumentation and 24 hours for those Functions that do not have channel components common to RPS instrumentation), has 3 and 4 been shown to be acceptable (Refs. ~~4 and 5~~) to permit restoration of any inoperable channel to OPERABLE status. This out of service time is only acceptable provided the associated Function is still maintaining isolation capability (refer to Required Action B.1 Bases). If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action A.1. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an isolation), Condition C must be entered and its Required Actions taken.

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated penetration flow path(s) or a complete loss of automatic initiation capability for the SGT System. A Function is considered to be maintaining isolation capability when sufficient channels are OPERABLE or in trip, such that one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIVs in the associated penetration flow path and one SGT subsystem can be initiated on an isolation signal from the given Function. For the Functions with two two-out-of-two logic trip systems (Functions 1, 2, and 3), this would require one trip system to have two channels, each OPERABLE or in trip. The Condition does not include the Manual Initiation Function (Function 4), since it is not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains isolation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Action(s) taken.

3 and 4

This Note is based on the reliability analysis (Refs. ~~2 and 5~~) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the SCIVs will isolate the associated penetration flow paths and the SGT System will initiate when necessary.

SR 3.3.6.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the indicated parameter for one instrument channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.2.4 (continued)

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

REFERENCES

1. FSAR, Section~~s~~ 15.6.5 ~~and 15.6.6.~~
~~2. FSAR, Section 15.7.4.~~
2 ~~3.~~ 10 CFR 50.36(c)(2)(ii).
3 ~~4.~~ NEDO-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.
4 ~~5.~~ NEDC-30851-P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

B 3.3 INSTRUMENTATION

B 3.3.7.1 Control Room Emergency Filtration (CREF) System Instrumentation

BASES

BACKGROUND

The CREF System is designed to provide a radiologically controlled environment to ensure the habitability of the control room for the safety of control room operators under all plant conditions. Two independent CREF subsystems are each capable of fulfilling the stated safety function. ~~Some~~ The instrumentation and controls for the CREF System automatically initiate action to pressurize the main control room (MCR) to minimize the consequences of radioactive material in the control room environment. ~~The other instrumentation (Main Control Room Ventilation Monitors) only provide alarm and indication in the control room to assist operators in the administrative control of the valves in the remote air intake plenums.~~

In the event of a loss of coolant accident (LOCA) signal (Reactor Vessel Water Level — Low Low, Level 2, Drywell Pressure — High, or Reactor Building Vent Exhaust Plenum Radiation — High), the CREF System is automatically started in the pressurization mode. Sufficient outside air is drawn in through two separate remote fresh air intakes to keep the MCR slightly pressurized with respect to the radwaste and turbine buildings. The outside air is then circulated through the charcoal filter. Both intakes are physically remote from all plant structures. ~~Redundant radiation monitors sensing the radiation level at each of the two remote intake headers are provided. The valves in the remote intake can be closed manually if the radiation level at the intake rises above an allowable level. Only one remote intake is closed at one time to maintain control room pressurization through one open remote intake.~~

The CREF System automatic initiation instrumentation has two trip systems: one trip system initiates one CREF subsystem, while the second trip system initiates the other CREF subsystem (Ref. 1). Each trip system receives input from the automatic initiation Functions listed above. Each of these Functions are arranged in a two-out-of-two logic for each trip system. The channels include electronic equipment (e.g., trip relays) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel outputs a CREF System initiation

(continued)

BASES

BACKGROUND (continued)

signal to the initiation logic. ~~The Main Control Room Ventilation Radiation Monitors only provide alarm and indication. The radiation monitors also include electronic equipment that compares measured input signals to pre-established setpoints. When the setpoint is exceeded, the radiation monitors output relay actuates, which then outputs to an alarm in the control room.~~

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The ability of the CREF System to maintain the habitability of the MCR is explicitly assumed for certain accidents as discussed in the FSAR safety analyses (Refs. 2 and 3). CREF System operation ensures that the radiation exposure of control room personnel, through the duration of any one of the postulated accidents, does not exceed the limits set by GDC 19 of 10 CFR 50, Appendix A~~.~~ and 10 CFR 50.67.

CREF instrumentation satisfies Criterion 3 of Reference 4.

The OPERABILITY of the CREF System instrumentation is dependent upon the OPERABILITY of the individual instrumentation channel Functions specified in Table 3.3.7.1-1. Each Function must have a required number of OPERABLE channels, with their setpoints within the specified Allowable Values, where appropriate. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions.

Allowable Values are specified for each CREF System Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. These nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint that is less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. Drywell Pressure – High

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). A high drywell pressure signal could indicate a LOCA and will automatically initiate the CREF System, since this could be a precursor to a potential radiation release and subsequent radiation exposure to control room personnel.

Drywell Pressure – High signals are initiated from four pressure switches that sense drywell pressure. Four channels of Drywell Pressure – High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System initiation.

The Drywell Pressure – High Allowable Value was chosen to be the same as the Secondary Containment Isolation Drywell Pressure – High Allowable Value (LCO 3.3.6.2).

The Drywell Pressure – High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. In MODES 4 and 5, the Drywell Pressure – High Function is not required since there is insufficient energy in the reactor to pressurize the drywell to the Drywell Pressure – High setpoint.

3. Reactor Building Vent Exhaust Plenum Radiation – High

~~NOTE~~

~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~

High secondary containment exhaust radiation is an indication of possible gross failure of the fuel cladding. The release may have originated from the primary containment due to a break in the RCPB or the refueling floor due to a fuel handling accident. When Reactor Building Vent Exhaust Plenum Radiation – High is detected, the CREF System is automatically initiated since this radiation release could result in radiation exposure to control room personnel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. Reactor Building Vent Exhaust Plenum Radiation—High (continued)

Reactor Building Vent Exhaust Plenum Radiation—High signals are initiated from four radiation monitors that measure radiation in the reactor building vent. Four channels of Reactor Building Vent Exhaust Plenum Radiation—High Function are available (two channels per trip system) and are required to be OPERABLE to ensure that no single instrument failure can preclude CREF System initiation.

The Reactor Building Vent Exhaust Plenum Radiation—High Allowable Value was chosen to be the same as the Secondary Containment Isolation Reactor Building Vent Exhaust Plenum Radiation—High Allowable Value (LCO 3.3.6.2).

The Reactor Building Vent Exhaust Plenum Radiation—High Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. The Function is also required to be OPERABLE during ~~CORE ALTERATIONS,~~ OPDRVs~~, and movement of irradiated fuel assemblies in the secondary containment~~ in case of fuel uncovery ~~or a fuel handling accident~~ that could cause a radioactive release to the environment.

~~4. Main Control Room Ventilation Radiation Monitor~~

~~The Main Control Room Ventilation Radiation Monitor measures radiation levels at the remote air intake plenums. A high radiation level may pose a threat to MCR personnel; thus a detector indicating this condition automatically initiates an alarm to alert MCR personnel.~~

~~Main Control Room Ventilation Radiation Monitor signals are initiated from four radiation monitors that measure radiation in the control room ventilation remote intake plenums. Four channels of Main Control Room Ventilation Radiation Monitor Function are available (two channels per remote intake plenum) and are required to be OPERABLE to alarm operators as to which Main Control Room Ventilation remote intake is in the potential radioactive plume generated from a design basis LOCA.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~4. Main Control Room Ventilation Radiation Monitor (continued)~~

~~The Allowable Value is selected to ensure protection of the MCR personnel.~~

~~The Main Control Room Ventilation Radiation Monitor Function is required to be OPERABLE in MODES 1, 2, and 3 to ensure that control room personnel are protected during a LOCA. The Function is also required to be OPERABLE during CORE ALTERATIONS, OPDRVs, and movement of irradiated fuel assemblies in the secondary containment in case of fuel uncovery or a fuel handling accident that could cause a radioactive release to the environment.~~

ℓ

ACTIONS

A Note has been provided to modify the ACTIONS related to CREF System instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable CREF System instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable CREF System instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.7.1-1. The applicable Condition specified in the Table is Function dependent. Each time an inoperable channel is discovered, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

With any Required Action and associated Completion Time of Condition B, C, or D not met, the associated CREF subsystem must be placed in the pressurization mode of operation (Required Action D.1) to ensure that control room personnel will be protected in the event of a Design Basis Accident. The method used to place the CREF subsystem in operation must provide for automatically reinitiating the subsystem upon restoration of power following a loss of power to the CREF subsystem(s). Alternately, if it is not desired to start the subsystem, the CREF subsystem associated with inoperable, untripped channels must be declared inoperable within 1 hour.

The 1 hour Completion Time is intended to allow the operator time to place the CREF subsystem in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, or for placing the associated CREF subsystem in operation.

~~E.1 and E.2~~

~~Because of the diversity of sensors available to provide radiation monitoring signals and the redundancy of the CREF System design, an allowable out of service time of 30 days has been provided to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided: a. the radiation monitoring capability is maintained for the associated remote air intake; and b. both channels associated with the other remote air intake are OPERABLE.~~

~~Radiation monitoring capability for a remote air intake is considered to be maintained when sufficient channels are OPERABLE to monitor the radiation at the remote air intake. This would require one channel to be OPERABLE at the remote air intake. In this situation (loss of radiation monitoring in a remote air intake), the 30 day allowance of Required Action E.2 is not appropriate without additional compensating actions. If radiation monitoring capability is not maintained at the associated remote air intake, the remote air intake must also be isolated within 1 hour of~~

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

discovery of loss of radiation monitoring capability at the remote air intake (Required Action E.1). This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action E.1, the Completion Time only begins upon discovery that both Main Control Room Ventilation Radiation Monitors on one remote air intake are inoperable. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoring of channels or isolating the remote air intake. If it is not desired to isolate the remote air intake (e.g., as in the case where the other remote air intake is already isolated), Condition F must be entered and its Required Actions taken. In addition, pursuant to LCO 3.0.6, the CREF System ACTIONS would not be entered even if both remote air intakes were isolated. Therefore, Required Action E.1 is modified by a Note to indicate that when both remote air intakes are isolated (due to complying with the Required Action E.1), ACTIONS for LCO 3.7.3, "Control Room Emergency Filtration (CREF) System," must be immediately entered. This allows Condition E to provide requirements for loss of one or more radiation monitoring channels without regard to whether both remote air intakes are isolated. LCO 3.7.3 provides the appropriate restrictions for both remote air intakes isolated.

With one or both channels associated with the other remote air intake inoperable, the 30 day allowance of Required Action E.2 is also not appropriate. In this situation (channels associated with both remote air intakes inoperable), there is a potential that a single failure can result in loss of radiation monitoring capability for both remote air intakes. Therefore, an allowable out of service time of 7 days from discovery of inoperable channels associated with both remote air intakes has been provided to restore all channels associated with one remote air intake to OPERABLE status. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For the first Completion Time of Required Action E.2, the Completion Time only begins upon discovery that one or more Main Control Room Ventilation

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

Radiation Monitors on both remote air intakes are inoperable. The 7 day Completion Time is based on the low probability of a DBA occurring during this time period, and is consistent with the time provided in the CREF System ACTIONS when one subsystem is inoperable (the monitors could be in a condition susceptible to a single failure that results in a loss of CREF System function, similar to when one subsystem is inoperable).

F.1

With any Required Action and associated Completion Time of Condition E not met, the radiation monitoring capability for one or both remote air intakes may be lost, therefore both CREF subsystems must be declared inoperable immediately.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each CREF System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are also modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREF System initiation or radiation monitoring capability, as applicable. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 ~~(Ref. 1)~~.

50.67

This LCO contains ~~iodine specific~~ activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to ~~a small fraction~~ 10% of the 10 CFR 100 limit.

50.67

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the FSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB outside containment during steady state operation, will not exceed ~~10% of the~~ dose guidelines of 10 CFR 100.

50.67

The MSLB analysis (Ref. 2) evaluates two source term cases. The source term for the first case is based on the Dose Equivalent I-131 limit of 0.2 μCi/gm provided in the LCO. The second case postulates a pre-accident iodine spike and uses a 4.0 μCi/gm Dose Equivalent I-131 source term. For the first case, the regulatory limit for the offsite dose is 10% of the limit specified in 10 CFR 50.67. The full offsite dose limit of 10 CFR 50.67 is applicable to the pre-accident iodine spiking case.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of Reference 3.

LCO

The specific iodine activity is limited to ≤ 0.2 µCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

50.67

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 µCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A Note permits the use of the provisions of LCO 3.0.4.c. This allowance permits entry into the applicable MODE(S) while relying on the ACTIONS.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

This allowance 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.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

50.67

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level. This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR ~~100.11~~. 50.67, "Accident Source Term."
2. FSAR, Section 15.6.4.
3. 10 CFR 50.36(c)(2)(ii).

BASES

BACKGROUND (continued)

This Specification ensures that the performance of the primary containment, in the event of a DBA, meets the assumptions used in the safety analyses of References 1 and 2. SR 3.6.1.1.1 leakage rate requirements are in conformance with 10 CFR 50, Appendix J, Option B (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

an inadequate core cooling event that degrades into core damage

The DBA that postulates the maximum release of radioactive material within primary containment is ~~a double-ended recirculation suction line break LOCA~~. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 38 psig (Ref. 4).

Primary containment satisfies Criterion 3 of Reference 5.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. In addition, the leakage from the drywell to the suppression chamber must be limited to ensure the pressure suppression function is accomplished and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

that component failures that might have affected this test are identified by other primary containment SRs. Two consecutive test failures, however, would indicate unexpected primary containment degradation; in this event, as the Note indicates, increasing the Frequency to once every 12 months is required until the situation is remedied as evidenced by passing two consecutive tests.

REFERENCES

1. FSAR, Section 6.2.1.1.3.
2. FSAR, Section 15.1.6 & 15.6.5
3. 10 CFR 50, Appendix J, Option B.
4. FSAR, Section 6.2.6.1.
5. 10 CFR 50.36(c)(2)(ii).

BASES

BACKGROUND (continued)

The 24 and 30 inch primary containment purge valves are PCIVs that are qualified for use during all operational conditions. The 24 and 30 inch primary containment purge valves are normally maintained closed in MODES 1, 2, and 3 to ensure the primary containment boundary is maintained. However, these purge valves may be open when being used for pressure control, inerting, de-inerting, ALARA, or air quality considerations since they are fully qualified. Two inch bypass lines with isolation valves bypass each primary containment purge valve when the 24 and 30 inch purge valves cannot be open.

APPLICABLE SAFETY ANALYSES

The PCIVs LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the primary containment boundary during major accidents. As part of the primary containment boundary, PCIV OPERABILITY supports leak tightness of primary containment. Therefore, the safety analysis of any event requiring isolation of primary containment is applicable to this LCO.

The DBAs that result in a release of radioactive material for which the consequences are mitigated by PCIVs are a loss of coolant accident (LOCA) and a main steam line break (MSLB) (Ref. 1). In the analysis for each of these accidents, it is assumed that PCIVs are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through PCIVs (including primary containment purge valves) are minimized. Of the events analyzed in Reference 1, the LOCA is the most limiting event due to radiological consequences. The closure time of the main steam isolation valves (MSIVs) is a significant variable from a radiological standpoint. The MSIVs are required to close within 3 to 5 seconds since the 3 second closure time is assumed in the MSIV closure (the most severe overpressurization transient) analysis (Ref. 2) and 5 second closure time is assumed in the MSLB analysis (Ref. 1). The safety analyses assume that the purge valves are closed at event initiation. Likewise, it is assumed that the primary containment isolates such that release of fission products to the environment is controlled.

(continued)

The radiological consequences associated with MSIV leakage following the design basis LOCA is based on the testing leakage limit of 16.0 scfh as specified in this surveillance. The test pressure, P_t, (25 psig) specified in this surveillance is less than the peak accident pressure, P_a. The specified P_t is less than P_a due to testing configuration constraints. The leakage assumed in the design basis LOCA analysis (Ref. 7) is calculated by converting the specified test leakage limit to the equivalent leakage rate for P_a conditions. This surveillance

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.11

~~The analyses in Reference 1 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ 11.5 scfh when tested at P_t (25 psig).~~ This ensures that MSIV leakage is properly accounted for in determining the overall primary containment leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.12

Surveillance of hydrostatically tested lines provides assurance that the calculation assumptions of Reference 1 are met. The acceptance criteria for the combined leakage of all hydrostatically tested lines is ≤ 1.0 gpm times the total number of hydrostatically tested PCIVs when tested at 1.1 P_a (41.8 psig). The combined leakage rates must be tested at the Frequency required by the Primary Containment Leakage Rate Testing Program.

REFERENCES

1. FSAR, Chapter 6.2.
2. FSAR, Section 15.2.4.
3. 10 CFR 50.36(c)(2)(ii).
4. Licensee Controlled Specifications Manual.
5. NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation," dated June 2000.
6. FSAR, Section 15.6.4.
7. FSAR, Section 15.6.5.
8. Regulatory Guide 1.183, Appendix A, July 2000.

The RHR drywell spray is operated post-LOCA to wash inorganic iodines and particulates from the drywell atmosphere into the suppression pool and to reduce primary containment pressure.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Residual Heat Removal (RHR) Drywell Spray

BASES

BACKGROUND

The primary containment is designed with a suppression pool so that, in the event of a loss of coolant accident (LOCA), steam released from the primary system is channeled through the suppression pool water and condensed without producing significant pressurization of the primary containment. The primary containment is designed so that with the pool initially at the minimum water volume and the worst single failure of the primary containment heat removal systems, suppression pool energy absorption combined with subsequent operator controlled pool cooling will prevent the primary containment pressure from exceeding its design value. However, the primary containment must also withstand a postulated bypass leakage pathway that allows the passage of steam from the drywell directly into the suppression pool airspace, bypassing the suppression pool. The RHR Drywell Spray System is designed to mitigate the effects of bypass leakage.

R

There are two redundant, 100% capacity RHR drywell spray subsystems. Each subsystem consists of a suction line from the suppression pool, an RHR pump, an RHR heat exchanger, and one spray sparger inside the drywell. Dispersion of the spray water is accomplished by spray nozzles in each subsystem.

The RHR drywell spray mode will be manually initiated, if required, following a LOCA, according to emergency procedures.

The RHR drywell spray is credited for two functions in the LOCA analysis (Ref. 3). The RHR drywell spray is credited for scrubbing inorganic iodines and particulates from the primary containment atmosphere. This function reduces the amount of airborne activity available for leakage from primary containment. The RHR drywell spray is also credited for primary containment pressure reduction. This function reduces the leak rate of airborne activity from primary containment.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum allowable bypass leakage area.

The equivalent flow path area for bypass leakage has been specified to be 0.05 ft^2. The analysis demonstrates that with drywell spray operation the primary containment pressure remains within design limits.

The RHR drywell spray satisfies Criterion 3 of Reference 2.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.5.2

This Surveillance is performed every 10 years to verify, by performance of an air or smoke flow test, that the spray nozzles are not obstructed and that flow will be provided when required. The 10 year Frequency is adequate to detect degradation in performance due to the passive nozzle design and its normally dry state and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Section 6.2.1.1.5.4.

2. 10 CFR 50.36(c)(2)(ii).

3. FSAR, Section 15.6.5

DELETE ENTIRE BASES SECTION B 3.6.1.8

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.8 Main Steam Isolation Valve Leakage Control (MSLC) System

BASES

BACKGROUND

The MSLC System supplements the isolation function of the MSIVs by processing the fission products that could leak through the closed MSIVs after a Design Basis Accident (DBA) loss of coolant accident (LOCA).

The MSLC System consists of two independent subsystems: an inboard subsystem, which is connected between the inboard and outboard MSIVs; and an outboard subsystem, which is connected to the main steam drain line header immediately downstream of the outboard MSIVs. Each subsystem is capable of processing leakage from MSIVs following a DBA LOCA. Each subsystem consists of a blower, valves, and piping. The inboard subsystem is also provided with four electric heaters to boil off any condensate prior to the gas mixture passing through the flow limiter.

Each subsystem operates in two process modes: depressurization and bleedoff. The depressurization process reduces the steam line pressure to within the operating capability of equipment used for the bleedoff mode. The effluent is discharged to the reactor building, which encloses a volume served by the Standby Gas Treatment (SGT) System. During bleedoff (long term leakage control), the blowers maintain a negative pressure in the main steam lines (Ref. 1). This ensures that leakage through the closed MSIVs is collected by the MSLC System. In this process mode, the effluent is discharged directly to the SGT System.

The MSLC System is manually initiated, and is not required to be initiated until the pressure of the steam trapped between the MSIVs decreases to the reactor steam dome pressure. The pressure requirement is estimated to take at least 1 hour (Ref. 1).

APPLICABLE SAFETY ANALYSES

The MSLC System mitigates the consequences of a DBA LOCA by ensuring that fission products that may leak from the closed MSIVs are filtered by the SGT System (Ref. 2). The analyses

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

BASES

APPLICABLE SAFETY ANALYSES (continued)	in Reference 3 provide the evaluation of offsite dose consequences. The operation of the MSLC System prevents a release of untreated leakage for this type of event. The MSLC System satisfies Criterion 3 of Reference 4.
LCO	One MSLC subsystem can provide the required processing of the MSIV leakage. To ensure that this capability is available, assuming worst case single failure, two MSLC subsystems must be OPERABLE.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could lead to a fission product release. Therefore, MSLC System OPERABILITY is required during these MODES. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the MSLC System OPERABLE is not required in MODE 4 or 5 to ensure MSIV leakage is processed.
ACTIONS	<u>A.1</u> With one MSLC subsystem inoperable, the inoperable MSLC subsystem must be restored to OPERABLE status within 30 days. In this condition, the remaining OPERABLE MSLC subsystem is adequate to perform the required leakage control function. However, the overall reliability is reduced because a single failure in the remaining subsystem could result in a total loss of MSIV leakage control function. The 30 day Completion Time is based on the redundant capability afforded by the remaining OPERABLE MSLC subsystem and the low probability of a DBA LOCA occurring during this period. <u>B.1</u> With two MSLC subsystems inoperable, at least one subsystem must be restored to OPERABLE status within 7 days. The 7 day Completion Time is based on the low probability of the occurrence of a DBA LOCA.

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

ACTIONS (continued)

C.1 and C.2

If the MSLC subsystem cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.8.1

Each MSLC System blower is operated for $\geq$ 15 minutes to verify OPERABILITY. The 31 day Frequency was developed considering the known reliability of the MSLC System blower and controls, the two subsystem redundancy, and the low probability of a significant degradation of the MSLC subsystem occurring between Surveillances and has been shown to be acceptable through operating experience.

SR 3.6.1.8.2

The electrical continuity of each inboard MSLC subsystem heater is verified by a resistance check, by verifying the rate of temperature increase meets specifications, or by verifying the current or wattage draw meets specifications. The 31 day Frequency is based on operating experience that has shown that these components usually pass this Surveillance when performed at this Frequency.

SR 3.6.1.8.3

A system functional test is performed to ensure that the MSLC System will operate through its operating sequence. This includes verifying that the automatic positioning of the valves and the operation of each interlock and timer are correct, that the blowers start and develop a flow rate of $\geq$ 24 cfm and $\leq$ 36 cfm, at a vacuum of $\geq$ 17 inches water gauge, and the upstream heaters meet current or wattage draw

(continued)

DELETE ENTIRE BASES SECTION B 3.6.1.8

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.1.8.3 (continued) requirements. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
REFERENCES	1. FSAR, Section 6.7.3. 2. FSAR, Section 6.7.2.1. 3. FSAR, Sections 15.6.5 and 15.F.6. 4. 10 CFR 50.36(c)(2)(ii).

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.1 Secondary Containment

BASES

BACKGROUND

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a Design Basis Accident (DBA). In conjunction with operation of the Standby Gas Treatment (SGT) System and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

The secondary containment is a structure that completely encloses the primary containment and those components that may be postulated to contain primary system fluid. This structure forms a control volume that serves to hold up and dilute the fission products. It is possible for the pressure in the control volume to rise relative to the environmental pressure (e.g., due to pump/motor heat load additions). To prevent ground level exfiltration while allowing the secondary containment to be designed as a conventional structure, the secondary containment requires support systems to maintain the control volume pressure at less than the external pressure. Requirements for these systems are specified separately in LCO 3.6.4.2, "Secondary Containment Isolation Valves (SCIVs)," and LCO 3.6.4.3, "Standby Gas Treatment (SGT) System."

APPLICABLE SAFETY ANALYSES

There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a loss of coolant accident (LOCA) (Ref. 1), and a fuel handling accident (Ref. 2). The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials from the primary containment is restricted to those leakage paths and

is

this

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

associated leakage rates assumed in the accident analysis, and that fission products entrapped within the secondary containment structure will be treated by the SGT System prior to discharge to the environment.

Secondary containment satisfies Criterion 3 of Reference ~~1~~ 2.

LCO

An OPERABLE secondary containment provides a control volume into which fission products that bypass or leak from primary containment, or are released from the reactor coolant pressure boundary components located in secondary containment, can be diluted and processed prior to release to the environment. For the secondary containment to be considered OPERABLE, it must have adequate leak tightness to ensure that the required vacuum can be established and maintained.

APPLICABILITY

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~

In MODES 1, 2, and 3, a LOCA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, secondary containment OPERABILITY is required during the same operating conditions that require primary containment OPERABILITY.

In MODES 4 and 5, the probability and consequences of the LOCA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining secondary containment OPERABLE is not required in MODE 4 or 5 to ensure a control volume, except for other situations for which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs)~~, during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment~~.

(continued)

BASES (continued)

ACTIONS

A.1

If secondary containment is inoperable, it must be restored to OPERABLE status within 4 hours. The 4 hour Completion Time provides a period of time to correct the problem that is commensurate with the importance of maintaining secondary containment during MODES 1, 2, and 3. This time period ensures that the probability of an accident (requiring secondary containment OPERABILITY) occurring during periods where secondary containment is inoperable is minimal.

B.1 and B.2

If the secondary containment cannot be restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C.3

Movement of irradiated fuel assemblies in the secondary containment, CORE ALTERATIONS, and OPDRVs can be postulated to cause fission product release to the secondary containment. In such cases, the secondary containment is the only barrier to release of fission products to the environment. CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended if the secondary containment is inoperable.

Suspension of these activities shall not preclude completing an action that involves moving a component to a safe position. Also, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

(continued)

if the secondary containment is inoperable

BASES

ACTIONS

C.1, C.2, and C.3 (continued)

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action C.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.1

normal operating

This SR ensures that the secondary containment boundary is sufficiently leak tight to preclude exfiltration under expected wind conditions. The 24 hour Frequency of this SR was developed based on operating experience related to secondary containment vacuum variations during the applicable MODES and the low probability of a DBA occurring between surveillances.

Maintaining the secondary containment vacuum at greater than 0.0 inch of vacuum water gauge supports the pre-accident conditions assumed in the secondary containment drawdown analysis (Ref.4).

Furthermore, the 24 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal secondary containment vacuum condition.

SR 3.6.4.1.2 and SR 3.6.4.1.3

Verifying that secondary containment equipment hatches and each inner access door or each outer access door in each access opening are closed ensures that the infiltration of outside air of such a magnitude as to prevent maintaining the desired negative pressure does not occur. Verifying that all such openings are closed provides adequate assurance that exfiltration from the secondary containment will not occur. SR 3.6.4.1.2 also requires equipment hatches to be sealed. In this application, the term "sealed" has no connotation of leak tightness. Maintaining

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1.2 and SR 3.6.4.1.3 (continued)

secondary containment OPERABILITY requires verifying all inner doors or all outer doors in the access opening are closed. However, each secondary containment access door is normally kept closed, except when the access opening is being used for entry and exit or when maintenance is being performed on an access. The 31 day Frequency for these SRs has been shown to be adequate based on operating experience, and is considered adequate in view of the other indications of door and hatch status that are available to the operator.

SR 3.6.4.1.4 ~~and SR 3.6.4.1.5~~

Replace with insert on the following page.

~~The SGT System exhausts the secondary containment atmosphere to the environment through appropriate treatment equipment. To ensure that all fission products are treated, SR 3.6.4.1.4 verifies that the SGT System will rapidly establish and maintain a pressure in the secondary containment that is less than the lowest postulated pressure external to the secondary containment boundary. This is confirmed by demonstrating that one SGT subsystem will draw down the secondary containment to ≥ 0.25 inches of vacuum water gauge in ≤ 120 seconds. This cannot be accomplished if the secondary containment boundary is not intact. SR 3.6.4.1.5 demonstrates that each SGT subsystem can maintain ≥ 0.25 inches of vacuum water gauge for 1 hour at a flow rate ≤ 2240 cfm. The 1 hour test period allows secondary containment to be in thermal equilibrium at steady state conditions. Therefore, these two tests are used to ensure secondary containment boundary integrity. Since these SRs are secondary containment tests, they need not be performed with each SGT subsystem. The SGT subsystems are tested on a STAGGERED TEST BASIS, however, to ensure that in addition to the requirements of LCO 3.6.4.3, either SGT subsystem will perform this test.~~ Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

INSERT FOR SR 3.6.4.1.4 BASIS

The inleakage limit of 2430 cfm specified in this surveillance is based upon the free volume of the secondary containment and corresponds to the flow rate that equates to one volume per day. The purpose of SR 3.6.4.1.4 is to provide assurance that the leakage rate is maintained within the limit of the SRP (Ref. 3) and the leakage assumption in the drawdown analysis. SR 3.6.4.1.4 demonstrates the ability of the SGT system to maintain at least a 0.25 inch vacuum water gauge in the secondary containment under steady state conditions. A 1 hour test period provides a reasonable period of time to establish steady state conditions. This surveillance serves to demonstrate secondary containment integrity. SR 3.6.4.1.4 together with SR 3.6.4.3.3 provide reasonable assurance that the secondary containment and the SGT system are capable of mitigating the design basis LOCA by drawing down the secondary containment within the 20 minute drawdown time credited in the LOCA analysis (Ref. 1).

Since SR 3.6.4.1.4 is a secondary containment integrity test, it does not need to be performed in conjunction with each performance of SR 3.6.4.3.3. SR 3.6.4.3.3 is performed on each SGT subsystem on a 24-month frequency. SR 3.6.4.1.4 is performed on a 24-month staggered test basis. This frequency ensures one performance of SR 3.6.4.1.4 every 24 months using a single SGT subsystem on an alternating basis.

BASES (continued)

REFERENCES

1. FSAR, Sections 15.6.5 ~~and 15.7.6~~.

~~2. FSAR, Section 15.7.4.~~

2 ~~3.~~ 10 CFR 50.36(c)(2)(ii).

3. NUREG 0800, Standard Review Plan, Section 6.2.3, "Secondary Containment Functional Design," Revision 2, dated July 1981.

4. FSAR, Section 6.2.3.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

BASES

BACKGROUND

The function of the SCIVs, in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) (Refs. 1 and 2). Secondary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that fission products that leak from primary containment following a DBA, that are released during certain operations when primary containment is not required to be OPERABLE, or that take place outside primary containment, are maintained within the secondary containment boundary.

The OPERABILITY requirements for SCIVs help ensure that an adequate secondary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), and blind flanges are considered passive devices. Isolation barrier(s) for the penetration are discussed in Reference 3.

Automatic SCIVs close on a secondary containment isolation signal to establish a boundary for untreated radioactive material within secondary containment following a DBA or other accidents.

Other penetrations are isolated by the use of valves in the closed position or blind flanges.

APPLICABLE SAFETY ANALYSES

The SCIVs must be OPERABLE to ensure the secondary containment barrier to fission product releases is established. The principal accidents for which the secondary containment boundary is required are a loss of coolant accident (Ref. 1) and a fuel handling accident (Ref. 2). The secondary containment performs no active function in response to each of these limiting events, but

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the boundary established by SCIVs is required to ensure that leakage from the primary containment is processed by the Standby Gas Treatment (SGT) System before being released to the environment.

Maintaining SCIVs OPERABLE with isolation times within limits ensures that fission products will remain trapped inside secondary containment so that they can be treated by the SGT System prior to discharge to the environment.

SCIVs satisfy Criterion 3 of Reference ~~4~~ 3.

LCO

SCIVs form a part of the secondary containment boundary. The SCIV safety function is related to control of offsite radiation releases resulting from DBAs.

The automatic power operated isolation valves are considered OPERABLE when their isolation times are within limits and the valves actuate on an automatic isolation signal. The valves covered by this LCO, along with their associated stroke times, are listed in Reference ~~5~~ 4.

The normally closed isolation valves or blind flanges are considered OPERABLE when manual valves are closed or open in accordance with appropriate administrative controls, automatic SCIVs are de-activated and secured in their closed position, and blind flanges are in place. These passive isolation valves or devices are listed in Reference ~~5~~ 4.

APPLICABILITY

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

In MODES 1, 2, and 3, a DBA could lead to a fission product release to the primary containment that leaks to the secondary containment. Therefore, OPERABILITY of SCIVs is required.

In MODES 4 and 5, the probability and consequences of these events are reduced due to pressure and temperature limitations in these MODES. Therefore, maintaining SCIVs

(continued)

BASES

APPLICABILITY (continued)

OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

The ACTIONS are modified by three Notes. The first Note allows penetration flow paths to be unisolated intermittently under administrative controls. These controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when the need for secondary containment isolation is indicated.

The second Note provides clarification that, for the purpose of this LCO, separate Condition entry is allowed for each penetration flow path. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable SCIV. Complying with the Required Actions may allow for continued operation, and subsequent inoperable SCIVs are governed by subsequent Condition entry and application of associated Required Actions.

The third Note ensures appropriate remedial actions are taken, if necessary, if the affected system(s) are rendered inoperable by an inoperable SCIV.

A.1 and A.2

In the event that there are one or more penetration flow paths with one SCIV inoperable, the affected penetration flow path(s) must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criteria are a closed and de-activated automatic SCIV, a closed manual valve, and a blind flange. For penetrations isolated in accordance with Required Action A.1, the device used to isolate the penetration should be the closest available device to

(continued)

BASES

ACTIONS

B.1 (continued)

considering the time required to isolate the penetration and the low probability of a DBA, which requires the SCIVs to close, occurring during this short time.

The Condition has been modified by a Note stating that Condition B is only applicable to penetration flow paths with two isolation valves. This clarifies that only Condition A is entered if one SCIV is inoperable in each of two penetrations.

C.1 and C.2

If any Required Action and associated Completion Time cannot be met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

D.1~~, D.2, and D.3~~

If any Required Action and associated Completion Time cannot be met, the plant must be placed in a condition in which the LCO does not apply. ~~If applicable, CORE ALTERATIONS and the movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable,~~ action must be immediately initiated to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action D.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3~~

(continued)

BASES

ACTIONS

D.1, D.2, and D.3 (continued)

would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1

This SR verifies each secondary containment isolation manual valve and blind flange that is required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the secondary containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification that those SCIVs in secondary containment that are capable of being mispositioned are in the correct position.

Since these SCIVs are readily accessible to personnel during normal unit operation and verification of their position is relatively easy, the 31 day Frequency was chosen to provide added assurance that the SCIVs are in the correct positions.

Two Notes have been added to this SR. The first Note applies to valves and blind flanges located in high radiation areas and allows them to be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these isolation devices, once they have been verified to be in the proper position, is low.

A second Note has been included to clarify that SCIVs that are open under administrative controls are not required to meet the SR during the time the SCIVs are open. These controls consist of stationing a dedicated operator at the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.2.1 (continued)

controls of the valve, who is in continuous communication with the control room. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

SR 3.6.4.2.2

Verifying the isolation time of each power operated and each automatic SCIV listed in Licensee Controlled Specification Table 1.6.4.2-1 is within limits is required to demonstrate OPERABILITY. The isolation time test ensures that the SCIV will isolate in a time period less than or equal to that assumed in the safety analyses. The Frequency of this SR is in accordance with the Inservice Testing Program.

SR 3.6.4.2.3

Verifying that each automatic SCIV closes on a secondary containment isolation signal is required to prevent leakage of radioactive material from secondary containment following a DBA or other accidents. This SR ensures that each automatic SCIV will actuate to the isolation position on a secondary containment isolation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. FSAR, Section~~s~~ 15.6.5 ~~and 15.F.6~~.
2. ~~FSAR, Section 15.7.4.~~

(continued)

BASES

REFERENCES (continued)

2. FSAR, Section 6.2.3.2.
3. 10 CFR 50.36(c)(2)(ii).
4. Licensee Controlled Specifications Manual.

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4.3 Standby Gas Treatment (SGT) System

BASES

BACKGROUND

The SGT System is required by 10 CFR 50, Appendix A, GDC 41, "Containment Atmosphere Cleanup" (Ref. 1). The function of the SGT System is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment.

The SGT System consists of two fully redundant subsystems, each with its own set of ductwork, dampers, charcoal filter train, and controls.

Each charcoal filter train consists of (components listed in order of the direction of the air flow):

a. A moisture separator;

b. Two electric heater banks (one primary and one backup);

c. A prefilter bank;

d. A high efficiency particulate air (HEPA) filter bank;

e. Two charcoal adsorber banks;

f. A second HEPA filter bank; and

g. Two centrifugal fans (one primary and one backup) each with inlet flow control vanes.

secondary containment

The sizing of the SGT System equipment and components ~~is~~ based on the results of an infiltration analysis, as well as an exfiltration analysis. The internal pressure of the ~~SGT System~~ boundary region is maintained at a negative pressure of 0.25 inch water gauge when the SGT system is in operation, which represents the internal pressure required to ensure zero exfiltration of air from the building ~~using the 95% meteorological data~~ under adverse weather conditions.

(continued)

BASES

BACKGROUND (continued)

The moisture separator is provided to remove entrained water in the air, while the electric heaters reduce the relative humidity of the airstream to less than 70% (Ref. 2). The prefilter removes large particulate matter, while the HEPA filter is provided to remove fine particulate matter and protect the charcoal from fouling. The charcoal adsorber removes gaseous elemental iodine and organic iodides, and the final HEPA filter is provided to collect any carbon fines exhausted from the charcoal adsorber.

The SGT System automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system. Following initiation, one fan per subsystem starts. SGT System flows are controlled automatically by modulating inlet vanes installed on the SGT fans.

APPLICABLE SAFETY ANALYSES

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident ~~and fuel handling accidents~~ (Ref~~s~~. 3 ~~and 4~~). ~~For all events analyzed,~~ the SGT System is ~~shown to be~~ automatically initiated to reduce, via filtration and adsorption, the radioactive material released to the environment.

The SGT System satisfies Criterion 3 of Reference ~~5~~ 4.

LCO

Following a DBA, a minimum of one SGT subsystem is required to maintain the secondary containment at a negative pressure with respect to the environment and to process gaseous releases. Meeting the LCO requirements for two OPERABLE subsystems ensures operation of at least one SGT subsystem in the event of a single active failure. In addition, only the primary electric heater bank and centrifugal fan are required for OPERABILITY of each SGT subsystem.

APPLICABILITY

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

(continued)

BASES

APPLICABILITY (continued)

In MODES 1, 2, and 3, a DBA could lead to a fission product release to primary containment that leaks to secondary containment. Therefore, SGT System OPERABILITY is required during these MODES.

In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the SGT System OPERABLE is not required in MODE 4 or 5, except for other situations under which significant releases of radioactive material can be postulated, such as during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS, or during movement of irradiated fuel assemblies in the secondary containment.

ACTIONS

A.1

With one SGT subsystem inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE SGT subsystem is adequate to perform the required radioactivity release control function. However, the overall system reliability is reduced because a single failure in the OPERABLE subsystem could result in the radioactivity release control function not being adequately performed. The 7 day Completion Time is based on consideration of such factors as the availability of the OPERABLE redundant SGT subsystem and the low probability of a DBA occurring during this period.

B.1 and B.2

If the SGT subsystem cannot be restored to OPERABLE status within the required Completion Time in MODE 1, 2, or 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

and C.2

ACTIONS (continued)

C.1, C.2.1, C.2.2, and C.2.3

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, when Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE SGT subsystem should be immediately placed in operation. This Required Action ensures that the remaining subsystem is OPERABLE, that no failures that could prevent automatic actuation will occur, and that any other failure would be readily detected.

An alternative to Required Action C.1 is to immediately suspend activities that represent a potential for releasing radioactive material to the secondary containment, thus placing the unit in a condition that minimizes risk. If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be a sufficient reason to require a reactor shutdown.

D.1

If both SGT subsystems are inoperable in MODE 1, 2, or 3, the SGT System may not be capable of supporting the required radioactive release control function. Therefore, actions are required to enter LCO 3.0.3 immediately.

(continued)

BASES

ACTIONS (continued)

E.1, E.2, and E.3

When two SGT subsystems are inoperable, if applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be immediately suspended. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, actions must be immediately initiated to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until OPDRVs are suspended.

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, Required Action E.1 has been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.3.1

Operating (from the control room) each SGT subsystem for ≥ 10 continuous hours ensures that both subsystems are OPERABLE and that all associated controls are functioning properly. It also ensures that blockage, fan or motor failure, or excessive vibration can be detected for corrective action. Operation with the heaters on (automatic heater cycling to maintain temperature) for ≥ 10 continuous hours every 31 days eliminates moisture on the adsorbers and HEPA filters. The 31 day Frequency was developed in consideration of the known reliability of fan motors and controls and the redundancy available in the system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.4.3.2

5

This SR verifies that the required SGT filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The SGT System filter tests are in accordance with Regulatory Guide 1.52 (Ref. 8). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specified test frequencies and additional information are discussed in detail in the VFTP.

SR 3.6.4.3.3

and achieves at least 4800 cfm of air flow within 2 minutes

This SR requires verification that each SGT subsystem starts upon receipt of an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.6.2, "Secondary Containment Isolation Instrumentation," overlaps this SR to provide complete testing of the safety function. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Verification of a subsystem's ability to obtain at least 4800 cfm of airflow within 2 minutes, in conjunction with the performance of SR 3.6.4.1.4, provides reasonable assurance that the SGT subsystem can achieve and maintain a vacuum in secondary containment within the 20 minute drawdown period credited in the design basis LOCA analysis (Ref. 3). The 2 minute acceptance criterion supports the bounding scenario assumed in the drawdown analysis that is based upon a loss of offsite power followed by a SGT start sequence that includes a failure of the lead (primary) fan to start. For this bounding start sequence, the lag (backup) fan will autostart following a short time delay.

SR 3.6.4.3.4

This SR requires verification that the primary SGT filter cooling recirculation valve can be opened and the primary fan started. This ensures that the ventilation mode of SGT System operation is available. While this Surveillance can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 41.

2. FSAR, Section 6.5.1.2.

3. FSAR, Section~~s~~ 15.6.5 ~~and 15.F.6~~.

~~4. FSAR, Section 15.7.4.~~

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).

5 ~~6.~~ Regulatory Guide 1.52, Rev. 2.

B 3.7 PLANT SYSTEMS

B 3.7.3 Control Room Emergency Filtration (CREF) System

BASES

BACKGROUND

The CREF System provides a radiologically controlled environment from which the unit can be safely operated following a Design Basis Accident (DBA).

The safety related function of the CREF System used to control radiation exposure consists of two independent and redundant high efficiency air filtration subsystems for treatment of outside supply air. Each subsystem consists of an electric heater, a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a filter unit fan, a control room recirculation fan, and the associated ductwork and dampers. The electric heater is used to limit the relative humidity of the air entering the filter train. Prefilters and HEPA filters remove particulate matter that may be radioactive. The charcoal adsorbers provide a holdup period for gaseous iodine, allowing time for decay.

The safety related CREF System is a standby system, but most of the ductwork is common to the Control Room Heating, Ventilation, and Air Conditioning (HVAC) System, which is operated to maintain the control room environment during normal operation. Upon receipt of the initiation signal(s) (indicative of conditions that could result in radiation exposure to control room personnel), the CREF System automatically switches to the pressurization mode of operation to prevent infiltration of contaminated air into the control room. A system of dampers isolates the control room (from the normal intake and exhaust), and control room outside air flow is redirected and processed through either of the two filter subsystems.

total effective dose equivalent (TEDE)

The CREF System is designed to maintain the control room environment for a 30 day continuous occupancy after a DBA, without exceeding a 5 rem ~~whole body~~ dose. ~~or its equivalent to any part of the body.~~ CREF System operation in maintaining the control room habitability is discussed in the FSAR, Sections 6.4.1 and 9.4.1 (Refs. 1 and 2, respectively).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

DBA LOCA analysis

The ability of the CREF System to maintain the habitability of the control room is an explicit assumption for the safety analyses presented in the FSAR, Chapters 6, 15 and 15 (Refs. 3 and 4, respectively). The pressurization mode of the CREF System is assumed to operate following a loss of coolant accident, main steam line break, fuel handling accident, and control rod drop accident. The radiological doses to control room personnel as a result of the various DBAs are summarized in Reference 4. No single active failure will cause the loss of outside or recirculated air from the control room.

The CREF System satisfies Criterion 3 of Reference 5.

LCO

Two redundant subsystems of the CREF System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in exceeding a dose of 5 rem to the control room operators in the event of a DBA.

The CREF System is considered OPERABLE when the individual components necessary to control operator exposure are OPERABLE in both subsystems. A subsystem is considered OPERABLE when its associated:

a. Filter unit fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow and are capable of performing their filtration functions;

c. Heater, ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained; and

d. Control room recirculation fan is OPERABLE.

In addition, the control room boundary must be maintained, including the integrity of the walls, floors, ceilings, ductwork, and access doors, such that the pressurization of SR 3.7.3.4 can be met. However, it is acceptable for access doors to be opened for normal control room entry and exit and not consider it to be a failure to meet the LCO. The

(continued)

BASES

LCO (continued)

LCO is modified by a Note allowing the control room boundary to be opened intermittently under administrative controls. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering and exiting the area. For other openings, these controls consist of stationing a dedicated individual at the opening who is in continuous communication with the control room. This individual will have a method to rapidly close the opening when a need for control room boundary integrity is required.

APPLICABILITY

---NOTE---
Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.

In MODES 1, 2, and 3, the CREF System must be OPERABLE to control operator exposure during and following a DBA, since the DBA could lead to a fission product release.

In MODES 4 and 5, the probability and consequences of a DBA are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the CREF System OPERABLE is not required in MODE 4 or 5, except for the following situations under which significant radioactive releases can be postulated:

a. During movement of irradiated fuel assemblies in the secondary containment;

b. During CORE ALTERATIONS; and

c. During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one CREF subsystem inoperable, the inoperable CREF subsystem must be restored to OPERABLE status within 7 days. With the unit in this condition, the remaining OPERABLE CREF subsystem is adequate to perform control room radiation

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, D.2.2, and D.2.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition D are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations.

Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, if the inoperable CREF subsystem cannot be restored to OPERABLE status within the required Completion Time, the OPERABLE CREF subsystem may be placed in the pressurization mode. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent automatic actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action D.1 is to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Actions must continue until the OPDRVs are suspended.

(continued)

BASES

ACTIONS (continued)

E.1

If both CREF subsystems are inoperable in MODE 1, 2, or 3, for reasons other than an inoperable control room boundary (i.e., Condition B) the CREF System may not be capable of performing the intended function and the unit is in a condition outside of the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

F.1, F.2, and F.3 ℓ

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition F are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs, with two CREF subsystems inoperable, action must be taken immediately to suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk. ℓ ℓ

If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe position. If applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.7.3.4

This SR verifies the integrity of the control room enclosure and the assumed inleakage rates of potentially contaminated air. The control room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper function of the CREF System. During the pressurization mode of operation, the CREF System is designed to slightly pressurize the control room to 0.125 inches water gauge positive pressure with respect to the radwaste and turbine buildings (as measured in the radwaste building cable spreading room) to prevent unfiltered inleakage. The CREF System is designed to maintain this positive pressure at an outside air flow rate of ≤ 1000 cfm through the control room in the pressurization mode. The Frequency of 24 months on a STAGGERED TEST BASIS is consistent with industry practice and other filtration system SRs.

REFERENCES

1. FSAR, Section 6.4.1.
2. FSAR, Section 9.4.1.
3. FSAR, Chapter 6.
4. FSAR, Chapter~~s~~ 15 ~~and 15.F~~.
5. 10 CFR 50.36(c)(2)(ii).
6. Regulatory Guide 1.52, Revision 2, March 1978.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Control Room AC System components are arranged in redundant safety related subsystems. During emergency operation, the Control Room AC System maintains a habitable environment and ensures the OPERABILITY of components in the control room. A single active failure of a component of the Control Room AC System, assuming a loss of offsite power, does not impair the ability of the system to perform its design function. Redundant detectors and controls are provided for control room temperature control when the emergency cooling coils are cooled by the Emergency Chilled Water System. The Control Room AC System is designed in accordance with Seismic Category I requirements. The Control Room AC System is capable of removing sensible and latent heat loads from the control room, including consideration of equipment heat loads and personnel occupancy requirements to ensure equipment OPERABILITY.

The Control Room AC System satisfies Criterion 3 of Reference 3.

LCO

Two independent and redundant subsystems of the Control Room AC System are required to be OPERABLE to ensure that at least one is available, assuming a single failure disables the other subsystem. Total system failure could result in the equipment operating temperature exceeding limits.

The Control Room AC System is considered OPERABLE when the individual components necessary to maintain the control room temperature are OPERABLE in both subsystems. These components include the emergency cooling coils (either cooled by the Emergency Chilled Water System or the SW System), control room recirculation fans, Emergency Chilled Water System chillers and pumps (if the Emergency Chilled Water System is being credited with providing cooling to the emergency cooling coils), ductwork, dampers, and associated instrumentation and controls. In addition, during conditions in MODES other than MODES 1, 2, and 3 when the Control Room AC System is required to be OPERABLE (e.g., ~~during CORE ALTERATIONS~~ OPDRVs), the necessary portions of the SW System and the ultimate heat sink are part of the OPERABILITY requirements covered by this LCO.

(continued)

BASES (continued)

APPLICABILITY

~~NOTE~~

~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~

In MODE 1, 2, or 3, the Control Room AC System must be OPERABLE to ensure that the control room temperature will not exceed equipment OPERABILITY limits following control room isolation.

In MODES 4 and 5, the probability and consequences of a Design Basis Accident are reduced due to the pressure and temperature limitations in these MODES. Therefore, maintaining the Control Room AC System OPERABLE is not required in MODE 4 or 5, except ~~for the following situations under which significant radioactive releases can be postulated:~~

~~a. During movement of irradiated fuel assemblies in the secondary containment;~~

~~b. During CORE ALTERATIONS; and~~

~~c.~~ During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

A.1

With one control room AC subsystem inoperable, the inoperable control room AC subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE control room AC subsystem is adequate to perform the control room air conditioning function. However, the overall reliability is reduced because a single failure in the OPERABLE subsystem could result in loss of the control room air conditioning function. The 30 day Completion Time is based on the low probability of an event occurring requiring control room isolation, the consideration that the remaining subsystem can provide the required protection, and the availability of alternate cooling methods.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

In MODE 1, 2, or 3, if the inoperable control room AC subsystem cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE that minimizes risk. To achieve this status the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1~~, C.2.1, C.2.2,~~ and C.2~~.3~~

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition C are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.~~

During ~~movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or~~ during OPDRVs, if Required Action A.1 cannot be completed within the required Completion Time, the OPERABLE control room AC subsystem may be placed immediately in operation. This action ensures that the remaining subsystem is OPERABLE, that no failures that would prevent actuation will occur, and that any active failure will be readily detected.

An alternative to Required Action C.1 is to immediately ~~suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.~~

~~If applicable, CORE ALTERATIONS and movement of irradiated fuel assemblies in the secondary containment must be suspended immediately. Suspension of these activities shall~~

(continued)

BASES

ACTIONS

C.1, C.2.1, C.2.2, and C.2.3 (continued)

not preclude completion of movement of a component to a safe position. Also, if applicable, action must be initiated immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

D.1

If both control room AC subsystems are inoperable in MODE 1, 2, or 3, the Control Room AC System may not be capable of performing the intended function. Therefore, LCO 3.0.3 must be entered immediately.

E.1, E.2, and E.3

LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the Required Actions of Condition E.1 are modified by a Note indicating that LCO 3.0.3 does not apply. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, inability to suspend movement of irradiated fuel assemblies is not sufficient reason to require a reactor shutdown.

During movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs with two control room AC subsystems inoperable, action must be taken to immediately suspend activities that present a potential for releasing radioactivity that might require isolation of the control room. This places the unit in a condition that minimizes risk.

If applicable, CORE ALTERATIONS and handling of irradiated fuel in the secondary containment must be suspended immediately. Suspension of these activities shall not preclude completion of movement of a component to a safe

(continued)

BASES

ACTIONS

E.1, E.2, and E.3 (continued)

~~position. Also, if applicable, action must be initiated~~ immediately to suspend OPDRVs to minimize the probability of a vessel draindown and subsequent potential for fission product release. Action must continue until the OPDRVs are suspended.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

This SR verifies that the heat removal capability of the system is sufficient to remove the control room heat load assumed in the safety analyses. The SR consists of a combination of testing and calculation. The 24 month Frequency is appropriate since significant degradation of the Control Room AC System is not expected over this time period.

REFERENCES

1. FSAR, Section 6.4.

2. FSAR, Section 9.4.1.

3. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the main condenser. Air and noncondensible gases are collected in the main condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensibles are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event as discussed in the FSAR, Section 11.3 (Ref. 1). The analysis assumes a single failure of a single component in the Main Condenser Offgas System. The gross gamma activity rate is controlled to ensure that during the event, the calculated offsite doses will be well within the limits (NUREG-0800, Ref. 2) of 10 CFR ~~100~~ (Ref. 3).
50.67

The main condenser offgas limits satisfy Criterion 2 of Reference 4.

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/Mwt-second after decay of 30 minutes. The LCO is established consistent with this requirement (3323 Mwt x 100 μCi/Mwt-second = 332 mCi/second) and is based on the original licensed RATED THERMAL POWER.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

This SR, on a 31 day Frequency, requires an isotopic analysis of an offgas sample (taken at the discharge of the main condenser air ejector prior to dilution) to ensure that the required limits are satisfied. The noble gases to be sampled are Xe-133, Xe-135, Xe-138, Kr-85, Kr-87, and Kr-88. If the measured rate of radioactivity increases significantly (by $\geq$ 50% after correcting for expected increases due to changes in THERMAL POWER), an isotopic analysis is also performed within 4 hours after the increase is noted, to ensure that the increase is not indicative of a sustained increase in the radioactivity rate. The 31 day Frequency is adequate in view of other instrumentation that continuously monitor the offgas, and is acceptable based on operating experience.

This SR is modified by a Note indicating that the SR is not required to be performed until 31 days after any main steam line is not isolated and the SJAE is in operation. Only in this condition can radioactive fission gases be in the Main Condenser Offgas System at significant rates.

REFERENCES

1. FSAR, Section 11.3.
2. NUREG-0800.
3. 10 CFR ~~100~~. 50.67 "Accident Source Term"
4. 10 CFR 50.36(c)(2)(ii).

B 3.7 PLANT SYSTEMS

B 3.7.7 Spent Fuel Storage Pool Water Level

BASES

BACKGROUND

The minimum water level in the spent fuel storage pool meets the assumptions of iodine decontamination factors following a fuel handling accident.

A general description of the spent fuel storage pool design is found in the FSAR, Section 9.1.2 (Ref. 1). The assumptions of the fuel handling accident are found in the FSAR, Section 15.7.4 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The water level above the irradiated fuel assemblies is an explicit assumption of the fuel handling accident (Ref. 2). A fuel handling accident is evaluated to ensure that the radiological consequences ~~(calculated whole body and thyroid doses at the exclusion area and low population zone boundaries) are ≤ 25% (NUREG/0800, Section 15.7.4, Ref. 3) of the 10 CFR 100 (Ref. 4) exposure guidelines.~~ are within the limits of 10 CFR 50.67 (Ref. 3) and the guidelines of RG 1.183 (Ref. 4). A fuel handling accident could release a fraction of the fission product inventory by breaching the fuel rod cladding as discussed in the Regulatory Guide ~~1.25~~ 1.183 (Ref. ~~5~~ 4).

The fuel handling accident is evaluated for the dropping of an irradiated fuel assembly onto the reactor core. The consequences of a fuel handling accident over the spent fuel storage pool are no more severe than those of the fuel handling accident over the reactor core (Ref. 2). The water level in the spent fuel storage pool provides for absorption of water soluble fission product gases and transport delays of soluble and insoluble gases that must pass through the water before being released to the secondary containment atmosphere. This absorption and transport delay reduces the potential radioactivity of the release during a fuel handling accident.

The spent fuel storage pool water level satisfies Criterion 2 of Reference ~~6~~ 5.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 9.1.2.

2. FSAR, Section 15.7.4.

~~3. NUREG-0800, Section 15.7.4, Revision 1, July 1981.~~

3 ~~4.~~ 10 CFR ~~100~~ 50.67, "Accident Source Term."

4 ~~5.~~ Regulatory Guide ~~1.25, March 1972~~ 1.183, July 2000.

5 ~~6.~~ 10 CFR 50.36(c)(2)(ii).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.2 AC Sources – Shutdown

BASES

BACKGROUND

A description of the AC sources is provided in the Bases for LCO 3.8.1, "AC Sources – Operating."

APPLICABLE SAFETY ANALYSES

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

The OPERABILITY of the minimum AC sources during MODES 4 and 5~~, and during movement of irradiated fuel assemblies in the secondary containment~~ ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate AC electrical power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel ~~or a fuel handling accident~~.

In general, when the unit is shutdown the Technical Specifications (TS) requirements ensure that the unit has the capability to mitigate the consequences of postulated accidents. However, assuming a single failure and concurrent loss of all offsite or loss of all onsite power is not required. The rationale for this is based on the fact that many Design Basis Accidents (DBAs), which are analyzed in MODES 1, 2, and 3, have no specific analyses in MODES 4 and 5. Worst case bounding events are deemed not credible in MODES 4 and 5 because the energy contained within the reactor pressure boundary, reactor coolant temperature and pressure, and the corresponding stresses result in the probabilities of occurrence significantly reduced or eliminated, and minimal consequences. These

(continued)

BASES (continued)

LCO

One offsite circuit supplying onsite Class 1E power distribution subsystem(s) of LCO 3.8.8, "Distribution Systems — Shutdown," ensures that all required loads are powered from offsite power. An OPERABLE DG, associated with a Division 1 or Division 2 Distribution System Engineered Safety Feature (ESF) bus required OPERABLE by LCO 3.8.8, ensures a diverse power source is available to provide electrical power support, assuming a loss of the offsite circuit. Similarly, when the high pressure core spray (HPCS) is required to be OPERABLE, an OPERABLE Division 3 DG ensures an additional source of power for the HPCS. Together, OPERABILITY of the required offsite circuit(s) and DG(s) ensures the availability of sufficient AC sources to operate the plant in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents, reactor vessel draindown).

The qualified offsite circuit(s) must be capable of maintaining rated frequency and voltage while connected to their respective ESF bus(es), and accepting required loads during an accident. Qualified offsite circuits are those that are described in the FSAR and are part of the licensing basis for the plant. The qualified offsite circuit includes the circuit path and disconnect to the respective transformer, the circuit path and breakers to the respective non-Class 1E 4.16 kV switchgear, SM-1, SM-2, and SM-3 (for the TR-S offsite circuit only), and the circuit path and breakers to the respective Class 1E switchgear (SM-4, SM-7, and SM-8) required by LCO 3.8.8.

The required DG must be capable of starting, accelerating to rated speed and voltage, and connecting to its respective ESF bus on detection of bus undervoltage, and accepting required loads. This sequence must be accomplished within 15 seconds for Divisions 1 and 2, and 18 seconds for Division 3. The DG-3 18 second start time includes the Loss of Voltage — Time Delay Function specified in LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation." Each DG must also be capable of accepting required loads within the assumed loading sequence intervals, and must continue to operate until offsite power can be restored to the ESF buses. These capabilities are required to be met from a variety of initial conditions such as: DG in standby with the engine hot and DG in standby with the engine at ambient conditions. Additional DG capabilities must be demonstrated to meet

(continued)

BASES

LCO (continued)

required Surveillances, e.g., capability of the DG to revert to standby status on an ECCS signal while operating in parallel test mode.

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY. The necessary portions of the Standby Service Water and HPCS Service Water systems are also required to provide appropriate cooling to each required DG.

It is acceptable for divisions to be cross tied during shutdown conditions, permitting a single offsite power circuit to supply all required divisions. No fast transfer capability is required for offsite circuits to be considered OPERABLE.

APPLICABILITY

The AC sources required to be OPERABLE in MODES 4 and 5 ~~and during movement of irradiated fuel assemblies in the secondary containment~~ provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

~~b. Systems needed to mitigate a fuel handling accident are available;~~

b ~~c~~. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c ~~d~~. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The AC power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.1.

ACTIONS

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel~~

(continued)

BASES

ACTIONS (continued)

assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1

An offsite circuit is considered inoperable if it is not available to one required ESF division. If two or more ESF 4.16 kV buses are required per LCO 3.8.8, division(s) with offsite power available may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for draining the reactor vessel. By the allowance of the option to declare required features inoperable that are not powered from offsite power, appropriate restrictions can be implemented in accordance with the required feature(s) LCOs' ACTIONS. Required features remaining powered from a qualified offsite power circuit, even if that circuit is considered inoperable because it is not powering other required features, are not declared inoperable by this Required Action.

and

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4

With the offsite circuit not available to all required divisions, the option still exists to declare all required features inoperable per Required Action A.1. Since this option may involve undesired administrative efforts, the allowance for sufficiently conservative actions is made. With the required DG inoperable, the minimum required diversity of AC power sources is not available. It is, therefore, required to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and activities that could potentially result in inadvertent draining of the reactor vessel.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize probability of the occurrence of postulated events. It is further required to initiate

(continued)

BASES

and

ACTIONS

A.2.1, A.2.2, A.2.3, A.2.4, B.1, B.2, B.3, and B.4
(continued)

action immediately to restore the required AC sources and to continue this action until restoration is accomplished in order to provide the necessary AC power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC electrical power sources should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

Pursuant to LCO 3.0.6, the Distribution System ACTIONS are not entered even if all AC sources to it are inoperable, resulting in de-energization. Therefore, the Required Actions of Condition A have been modified by a Note to indicate that when Condition A is entered with no AC power to any required ESF bus, ACTIONS for LCO 3.8.8 must be immediately entered. This Note allows Condition A to provide requirements for the loss of the offsite circuit whether or not a division is de-energized. LCO 3.8.8 provides the appropriate restrictions for the situation involving a de-energized division.

C.1

When the HPCS is required to be OPERABLE, and the Division 3 DG is inoperable, the required diversity of AC power sources to the HPCS is not available. Since these sources only affect the HPCS, the HPCS is declared inoperable and the Required Actions of LCO 3.5.2, "Emergency Core Cooling System – Shutdown" entered.

In the event all sources of power to Division 3 are lost, Condition A will also be entered and direct that the ACTIONS of LCO 3.8.8 be taken. If only the Division 3 DG is inoperable, and power is still supplied to HPCS, 72 hours is allowed to restore the DG to OPERABLE. This is reasonable considering HPCS will still perform its function, absent an additional single failure.

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.5 DC Sources — Shutdown

BASES

BACKGROUND

A description of the DC sources is provided in the Bases for LCO 3.8.4, "DC Sources — Operating."

APPLICABLE SAFETY ANALYSES

~~-----NOTE-----~~
~~Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.~~
~~----------~~

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapters 15 ~~and 15.F~~ (Ref. 2), assume that Engineered Safety Feature systems are OPERABLE. The DC electrical power system provides normal and emergency DC electrical power for the diesel generators, emergency auxiliaries, and control and switching during all MODES of operation ~~and during movement of irradiated fuel assemblies in the secondary containment~~.

The OPERABILITY of the DC subsystems is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum DC electrical power sources during MODES 4 and 5 ~~and during movement of irradiated fuel assemblies in the secondary containment~~ ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

(continued)

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a postulated inadvertent draindown of the vessel during shutdown.

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Adequate DC electrical power is provided to mitigate ~~events postulated during shutdown, such as an inadvertent draindown of the vessel or a fuel handling accident.~~

The DC sources satisfy Criterion 3 of Reference 3.

LCO

The DC electrical power subsystems, each consisting of one battery, one battery charger, and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the division, are required to be OPERABLE to support required Distribution System divisions required OPERABLE by LCO 3.8.8, "Distribution Systems – Shutdown." This ensures the availability of sufficient DC electrical power sources to operate the unit in a safe manner and to mitigate the consequences of ~~postulated events during shutdown (e.g., fuel handling accidents and inadvertent reactor vessel draindown).~~

APPLICABILITY

The DC electrical power sources required to be OPERABLE in MODES 4 and 5 ~~and during movement of irradiated fuel assemblies in the secondary containment~~ provide assurance that:

a. Required features to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies in the core in case of an inadvertent draindown of the reactor vessel;

~~b. Required features needed to mitigate a fuel handling accident are available;~~

b ~~c~~. Required features necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c ~~d~~. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

The DC electrical power requirements for MODES 1, 2, and 3 are covered in LCO 3.8.4.

(continued)

BASES (continued)

ACTIONS

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.~~

A.1, A.2.1, and A.2.2~~, A.2.3, and A.2.4~~

If more than one DC distribution subsystem is required according to LCO 3.8.8, the DC electrical power subsystems remaining OPERABLE with one or more DC electrical power subsystems inoperable may be capable of supporting sufficient required features to allow continuation of ~~CORE ALTERATIONS, fuel movement, and~~ operations with a potential for draining the reactor vessel. By allowing the option to declare required features inoperable with associated DC electrical power subsystem(s) inoperable, appropriate restrictions are implemented in accordance with the affected system LCOs' ACTIONS. However, in many instances this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend ~~CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment, and~~ any activities that could result in inadvertent draining of the reactor vessel).

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required DC electrical power subsystems and to continue this action until restoration is accomplished in order to provide the necessary DC electrical power to the plant safety systems.

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required DC electrical power subsystems

(continued)

BASES

ACTIONS

A.1, A.2.1, and A.2.2~~, A.2.3, and A.2.4~~ (continued)

should be completed as quickly as possible in order to minimize the time during which the plant safety systems may be without sufficient power.

SURVEILLANCE REQUIREMENTS

SR 3.8.5.1

SR 3.8.5.1 requires performance of all Surveillances required by SR 3.8.4.1 through SR 3.8.4.8. Therefore, see the corresponding Bases for LCO 3.8.4 for a discussion of each SR.

This SR is modified by a Note. The reason for the Note is to preclude requiring OPERABLE DC sources from being discharged below their capability to provide the required power supply or otherwise rendered inoperable during performance of SRs. It is the intent that these SRs must still be capable of being met, but actual performance is not required.

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapter~~s~~ 15 ~~and 15.F~~.
3. 10 CFR 50.36(c)(2)(ii).

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 Distribution Systems — Shutdown

BASES

BACKGROUND

A description of the AC and DC electrical power distribution systems is provided in the Bases for LCO 3.8.7, "Distribution Systems — Operating."

APPLICABLE SAFETY ANALYSES

-----------------------------------NOTE-----------------------------------
Handling a cask/canister loaded with spent fuel, after the canister is seal welded and leak tested, is not considered to be movement of irradiated fuel.

The initial conditions of Design Basis Accident and transient analyses in the FSAR, Chapter 6 (Ref. 1) and Chapters 15 and 15.F (Ref. 2), assume Engineered Safety Feature (ESF) systems are OPERABLE. The AC and DC electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC and DC electrical power distribution system is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC and DC electrical power sources and associated power distribution subsystems during MODES 4 and 5, and during movement of irradiated fuel assemblies in the secondary containment ensures that:

a. The facility can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

c. Adequate power is provided to mitigate events postulated during shutdown, such as an inadvertent draindown of the vessel ~~or a fuel handling accident~~.

The AC and DC electrical power distribution systems satisfy Criterion 3 of Reference 3.

LCO

Various combinations of subsystems, equipment, and components are required OPERABLE by other LCOs, depending on the specific plant condition. Implicit in those requirements is the required OPERABILITY of necessary support features. This LCO explicitly requires energization of the portions of the electrical distribution system necessary to support OPERABILITY of Technical Specifications' required systems, equipment, and components – both specifically addressed by their own LCOs, and implicitly required by the definition of OPERABILITY.

In addition, it is acceptable for required buses to be cross-tied during shutdown conditions, permitting a single source to supply multiple redundant buses, provided the source is capable of maintaining proper frequency (if required) and voltage.

Maintaining these portions of the distribution system energized ensures the availability of sufficient power to operate the plant in a safe manner to mitigate the consequences of postulated events during shutdown (e.g., ~~fuel handling accidents and~~ inadvertent reactor vessel draindown).

APPLICABILITY

The AC and DC electrical power distribution subsystems required to be OPERABLE in MODES 4 and 5 ~~and during movement of irradiated fuel assemblies in the secondary containment~~ provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

~~b. Systems needed to mitigate a fuel handling accident are available;~~

(continued)

BASES

APPLICABILITY (continued)

b. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

c. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown or refueling condition.

The AC and DC electrical power distribution subsystem requirements for MODES 1, 2, and 3 are covered in LCO 3.8.7.

ACTIONS

~~LCO 3.0.3 is not applicable while in MODE 4 or 5. However, since irradiated fuel assembly movement can occur in MODE 1, 2, or 3, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving irradiated fuel assemblies while in MODE 4 or 5, LCO 3.0.3 would not specify any action. If moving irradiated fuel assemblies while in MODE 1, 2, or 3, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.~~

and

A.1, A.2.1, A.2.2, ~~A.2.3, A.2.4, and A.2.5~~

Although redundant required features may require redundant divisions of electrical power distribution subsystems to be OPERABLE, one OPERABLE distribution subsystem division may be capable of supporting sufficient required features to allow continuation of ~~CORE ALTERATIONS, fuel movement, and~~ operations with a potential for draining the reactor vessel. By allowing the option to declare required features associated with an inoperable distribution subsystem inoperable, appropriate restrictions are implemented in accordance with the affected distribution subsystem LCO's Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend ~~CORE ALTERATIONS, movement of irradiated fuel assemblies in the secondary containment and~~ any activities that could result in inadvertent draining of the reactor vessel).

(continued)

BASES

ACTIONS

and &

A.1, A.2.1, A.2.2, A.2.3, A.2.4, and A.2.5 (continued)

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC and DC electrical power distribution subsystems and to continue this action until restoration is accomplished in order to provide the necessary power to the plant safety ~~systems.~~

and A.2.2

Notwithstanding performance of ~~the above conservative~~ Required Actions, a required residual heat removal-shutdown cooling (RHR-SDC) subsystem may be inoperable. In this case, Required Actions A.2.1 ~~through A.2.4~~ do not adequately address the concerns relating to coolant circulation and heat removal. Pursuant to LCO 3.0.6, the RHR-SDC ACTIONS would not be entered. Therefore, Required Action A.2.5 is provided to direct declaring RHR-SDC inoperable, which results in taking the appropriate RHR-SDC ACTIONS. 3

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required distribution subsystems should be completed as quickly as possible in order to minimize the time the plant safety systems may be without power.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the AC and DC electrical power distribution subsystems are functioning properly, with the correct breaker alignment. The correct breaker alignment ensures power is available to each required bus. The verification of energization of the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. This may be performed by verification of absence of low voltage alarms or by verifying a load powered from the bus is operating. The 7 day Frequency takes into account the redundant capability of the electrical power distribution subsystems, as well as other indications available in the control room that alert the operator to subsystem malfunctions.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Chapter 6.
2. FSAR, Chapters 15 and 15.F.
3. 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.6 Reactor Pressure Vessel (RPV) Water Level — Irradiated Fuel

BASES

BACKGROUND

The movement of irradiated fuel assemblies within the RPV requires a minimum water level of 22 ft above the top of the RPV flange. During refueling, this maintains a sufficient water level in the reactor vessel cavity and spent fuel storage pool. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ≤ 25% of 10 CFR 100 limits, as provided by the guidance of Reference 3.

≤ 6.3 rem TEDE

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).

Appendix B of RG 1.183

Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft (a decontamination factor of 100 is still expected at a water level as low as 22 ft) and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4). While the worst case assumptions include the dropping of the irradiated fuel assembly being handled onto the reactor core, the possibility exists of the dropped assembly striking the RPV flange and releasing fission products. Therefore, the minimum depth for water coverage to ensure acceptable radiological consequences is specified from the RPV flange. Since the worst case event

Replace with insert on following page

(continued)

INSERT FOR B.3.9.6 APPLICABLE SAFETY ANALYSIS

The 22 feet above the top of the RPV flange equates to approximately 52 feet above the fuel seated in the vessel. The analyzed fuel drop is assumed to occur in the reactor vessel cavity, as a drop from this location would create the bounding amount of fuel damage. The source term for this accident is the fission product inventory contained in the gap of the damaged rods. The fraction of fission product inventory assumed in the gap is specified in Table 3 of RG 1.183 (Ref. 1). Analysis of the FHA is described in Reference 2. The number of rods damaged includes rods from the dropped bundle and rods from impacted bundles seated in the vessel. An unobstructed drop over the reactor cavity results in the greatest amount of kinetic energy and the bounding amount of rod damage. A bundle dropped over the spent fuel pool or onto the vessel flange would result in reduced releases of fission gases.

A minimum water level of 23 feet above the fuel seated in the vessel allows an overall decontamination factor of 200 for the iodine released from the damaged rods (Appendix B of Ref. 1). With the minimum water level of 22 feet above the RPV flange and a minimum decay time of 24 hours prior to fuel movement, the analysis demonstrates that the resulting radiological consequences are within the allowable limits (Ref. 1 and 3).

BASES

APPLICABLE SAFETY ANALYSES (continued)	~~results in failed fuel assemblies seated in the core, as well as the dropped assembly, dropping an assembly on the RPV flange will result in reduced releases of fission gases.~~ RPV water level satisfies Criterion 2 of Reference ~~3~~ 4.
LCO	A minimum water level of 22 ft above the top of the RPV flange is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~3~~ 1.
APPLICABILITY	LCO 3.9.6 is applicable when moving irradiated fuel assemblies within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for handling of new fuel assemblies or control rods (where water depth to the RPV flange is not of concern) are covered by LCO 3.9.7, "RPV Water Level — New Fuel or Control Rods." Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level."
ACTIONS	A.1 If the water level is < 22 ft above the top of the RPV flange, all operations involving movement of irradiated fuel assemblies within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of irradiated fuel movement shall not preclude completion of movement of a component to a safe position.
SURVEILLANCE REQUIREMENTS	SR 3.9.6.1 Verification of a minimum water level of 22 ft above the top of the RPV flange ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is met. Water at the required level limits the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.6.1 (continued)

consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972~~ 1.183, July 2000
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4~~

3 ~~4.~~ 10 CFR ~~100.11~~ 50.67, "Accident Source Term."

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).

B 3.9 REFUELING OPERATIONS

B 3.9.7 Reactor Pressure Vessel (RPV) Water Level — New Fuel or Control Rods

BASES

BACKGROUND

The movement of new fuel assemblies or handling of control rods within the RPV when fuel assemblies seated within the reactor vessel are irradiated requires a minimum water level of ~~22~~ 23 ft above the top of irradiated fuel assemblies seated within the RPV. During refueling, this maintains a sufficient water level above the irradiated fuel. Sufficient water is necessary to retain iodine fission product activity in the water in the event of a fuel handling accident (Refs. 1 and 2). Sufficient iodine activity would be retained to limit offsite doses from the accident to ~~< 25% of 10 CFR 100 limits~~ ≤ 6.3 rem TEDE, as provided by the guidance of Reference ~~2~~ 1.

APPLICABLE SAFETY ANALYSES

During movement of new fuel assemblies or handling of control rods over irradiated fuel assemblies, the water level in the RPV is an initial condition design parameter in the analysis of a fuel handling accident in containment postulated by ~~Regulatory Guide 1.25 (Ref. 1). A minimum water level of 23 ft (Regulatory Position C.1.c of Ref. 1) allows a decontamination factor of 100 (Regulatory Position C.1.g. of Ref. 1) to be used in the accident analysis for iodine. This relates to the assumption that 99% of the total iodine released from the pellet to cladding gap of all the dropped fuel assembly rods is retained by the refueling cavity water. The fuel pellet to cladding gap is assumed to contain 10% of the total fuel rod iodine inventory (Ref. 1).~~

RG 1.183 (Ref. 1). A minimum water level of 23 feet above the fuel seated in the RPV allows an overall decontamination factor (DF) of 200 for the iodine released from the damaged rods. This DF is used in the Fuel Handling Accident (FHA) analysis (Ref. 2). The source term for this accident is the fission product inventory contained in the gap of the damaged rods. The fraction of fission product inventory assumed to be in the gap is specified in Table 3 of Regulatory Guide 1.183 (Ref. 1).

~~Analysis of the fuel handling accident inside containment is described in Reference 2. With a minimum water level of 22 ft (a decontamination factor of 100 is still expected at a water level as low as 22 ft) and a minimum decay time of 24 hours prior to fuel handling, the analysis and test programs demonstrate that the iodine release due to a postulated fuel handling accident is adequately captured by the water, and that offsite doses are maintained within allowable limits (Ref. 4).~~

With a minimum water level of 23 feet above the fuel seated in the RPV and a minimum decay time of 24 hours prior to fuel handling, the analysis demonstrates that the resulting radiological consequences are within the allowable limits (Ref. 1 and 3).

The related assumptions include the worst case dropping of an irradiated fuel assembly onto the reactor core loaded with irradiated fuel assemblies.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RPV water level satisfies Criterion 2 of Reference ~~1~~ 4.

LCO

A minimum water level of ~~22~~ 23 ft above the top of irradiated fuel assemblies seated within the RPV is required to ensure that the radiological consequences of a postulated fuel handling accident are within acceptable limits, as provided by the guidance of Reference ~~2~~ 1.

APPLICABILITY

LCO 3.9.7 is applicable when moving new fuel assemblies or handling control rods (i.e., movement with other than the normal control rod drive) over irradiated fuel assemblies seated within the RPV. The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. If irradiated fuel is not present within the RPV, there can be no significant radioactivity release as a result of a postulated fuel handling accident. Requirements for fuel handling accidents in the spent fuel storage pool are covered by LCO 3.7.7, "Fuel Pool Water Level." Requirements for handling irradiated fuel over the RPV are covered by LCO 3.9.6, "Reactor Pressure Vessel (RPV) Water Level – Irradiated Fuel."

ACTIONS

A.1

If the water level is < ~~22~~ 23 ft above the top of irradiated fuel assemblies seated within the RPV, all operations involving movement of new fuel assemblies and handling of control rods within the RPV shall be suspended immediately to ensure that a fuel handling accident cannot occur. The suspension of fuel movement and control rod handling shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

Verification of a minimum water level of ~~22~~ 23 ft above the top of the irradiated fuel assemblies seated within the RPV ensures that the design basis for the postulated fuel handling accident analysis during refueling operations is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1 (continued)

met. Water at the required level limits the consequences of damaged fuel rods, which are postulated to result from a fuel handling accident in containment (Ref. 2).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls on valve positions, which make significant unplanned level changes unlikely.

REFERENCES

1. Regulatory Guide ~~1.25, March 23, 1972.~~ 1.183, July 2000.
2. FSAR, Section 15.7.4.
3. ~~NUREG-0800, Section 15.7.4.~~

3 ~~4.~~ 10 CFR ~~100.11~~ 50.67, "Accident Source Term".

4 ~~5.~~ 10 CFR 50.36(c)(2)(ii).

New

B 3.9 REFUELING OPERATIONS

B 3.9.10 Decay Time

BASES

BACKGROUND

The postulated fuel handling accident involves the drop of a fuel assembly on top of the reactor core during refueling operations (Ref. 1). The drop over the reactor core is more limiting than the drop over the spent fuel pool since the kinetic energy for the drop over the reactor core area (greater than 23 feet) produces a larger number of damaged fuel pins on impact than the shorter drops that could occur over the fuel pool. The fuel handling accident is analyzed using Alternative Source Term methodology governed by 10 CFR 50.67 (Ref. 2) and the guidelines of Regulatory Guide 1.183 (Ref. 3).

The fuel handling accident analysis assumes that the accident occurs at least 24 hours after plant shutdown. Specifically, a 24-hour radioactive decay time of the fission product inventory is assumed during the interval between shutdown and movement of assemblies in the reactor core.

APPLICABLE SAFETY ANALYSES

The minimum requirement of 24 hours of reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is an initial condition of the fuel handling accident analysis.

Decay time satisfies the requirements of Criterion 2 of Reference 4.

LCO

The specified decay time limit requires the reactor to be subcritical for at least 24 hours. Implicit in this TS is the Applicability (during movement of irradiated fuel in the reactor vessel). This ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products, thus reducing the fission product inventory and reducing the effects of a fuel handling accident.

(continued)

New

BASES (continued)

APPLICABILITY

This decay time restriction is applicable only during movement of irradiated fuel in the reactor vessel following reactor operation. Therefore, it effectively prohibits movement of irradiated fuel in the reactor vessel during the first 24 hours following reactor shutdown.

ACTIONS

A.1

With the reactor subcritical less than 24 hours, all movement of irradiated fuel in the reactor vessel must be suspended. As stated above, movement of irradiated fuel in the reactor vessel is prohibited during the first 24 hours following reactor shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.9.10.1

Since movement of irradiated fuel in the reactor vessel is prohibited during the first 24 hours following reactor shutdown, a verification of time subcritical must be made prior to movement of irradiated fuel in the reactor vessel. This is done by confirming the time and date of subcriticality, and verifying that at least 24 hours have elapsed. The Frequency of "once prior to movement of irradiated fuel in the reactor vessel" ensures that the operation within the design basis assumption for decay time in the fuel handling accident analysis.

REFERENCES

1. FSAR, Section 15.7.4.
2. 10 CFR 50.67, "Accident Source Term."
3. Regulatory Guide 1.183, July 2000.
4. 10 CFR 50.36(c)(2)(ii).

Non-proprietary versions of Supporting Calculations

1. Energy Northwest Calculation NE-02-04-01, "Dose Calculation Database" Revision 2, dated September 23, 2004
2. Energy Northwest Calculation NE-02-03-14, "Control Room X/Q Using ARCON96 with the 1996-1999 Meteorological Data" Revision 0, dated June 22, 2004
3. Energy Northwest Calculation NE-02-04-07, "Control Rod Drop Accident Offsite and Control Room Doses" Revision 0, dated August 11, 2004
4. Energy Northwest Calculation NE-02-04-08, "Columbia Fuel Handling Accident Offsite and Control Room Doses Using Regulatory Guide 1.183 Source Terms" Revision 0, dated August 2, 2004
5. Energy Northwest Calculation NE-02-04-05, "Columbia Offsite and Control Room Doses for LOCA using AST and NRC Methods" Revision 0, dated August 4, 2004
6. Energy Northwest Calculation NE-02-04-06, "Main Steamline Break Accident Off-site and Control Room Doses" Revision 1, dated September 30, 2004
7. Energy Northwest Calculation NE-02-03-16, "Calculation of the EAB and LPZ X/Q values using PAVAN with the 1996 - 1999 Meteorological Data" Revision 0, dated May 30, 2004
8. Energy Northwest Calculation NE-02-03-15, "POST-LOCA SUPPRESSION POOL pH" Revision 0, dated August 3, 2004
9. Energy Northwest Calculation NE-02-01-05, "Secondary Containment Drawdown" Revision 1, dated September 30, 2004
10. CD-ROM* containing 1) 51-5029820-02, Columbia Generating Station, Meteorological Data Documents 1996-1999, 2) 32-5031898-01, Columbia Generating Station, Meteorological Data Input Document for ARCON96, and 3) 32-5032044-01, Columbia Generating Station, Meteorological Data Input Documents for PAVAN, CD-Rom dated October 2004
11. CD-ROM* containing 1) 51-5029820-02, Columbia Generating Station, Meteorological Data Files 1996-1999, 2) 32-5031898-01, Columbia Generating Station, Meteorological Data Input Files for ARCON96, and 3) 32-5032044-01, Columbia Generating Station, Meteorological Data Input Files for PAVAN, CD-Rom dated May 3, 2004

* CD-ROMs of electronic data files are enclosed with the Document Control Desk copy only.

List of Regulatory Commitments

Page 5

The updating of the FSAR to reflect these changes will be performed as part of the implementation of the LAR

Page 42

New procedural guidance is required to address reliance on SLC for pH control. The appropriate procedural guidance will be established during the implementation of the LAR. (See section 4.8.1 for additional information on the SLC system and the justification for the use of SLC in this application.)

Page 48

To preclude this undesirable operator action, the appropriate plant procedure(s) will be revised to prohibit the securing of a CREF train within the first 10 hours of the design basis LOCA.

Page 71

New procedural guidance is required to address reliance on SLC for pH control. The appropriate procedural guidance will be established during the implementation of the LAR. (See section 4.8.1 for additional information on the SLC system and the justification for the use of SLC in this application.)

Page 72

The TSGs will be revised to require manual initiation of the SLC system, at a level of 14,000 R/hr, and to continue injection until the SLC tank low level alarm is received.

Page 72

In addition, Technical Support Center (TSC) Operations Managers will receive training on the TSG revisions as part of the implementation of the approved AST changes.

Page 73

In addition, the changes to the TSGs for high containment radiation will instruct the operators to inject until low tank signal is received.