Enclosure

EVALUATION OF PROPOSED CHANGES

Subject: Modification of the Control Room Emergency Air Treatment System (CREATS), changing of the Dose Calculation Methodology to Alternate Source Term (10CFR50.67) and revision of Ginna Technical Specification Sections 1.1, 3.3.6, 3.4.16, 3.6.6, 3.7.9,5.5.10, 5.5.16 and 5.6.7.

1.0 DESCRIPTION

This letter is a request to amend Operating License Docket # 50-244 for the R.E. Ginna Nuclear Power Plant.

The proposed amendment would revise the operating license as follows:

- Reflect the new Control Room Emergency Air Treatment System (CREATS)
- Implement new industry guidance related to CREATS
- Add the requirement for a Control Room Integrity Program
- Add reporting requirements for an inoperable boundary
- Eliminate the requirement for Containment Post Accident Charcoal Filters
- Modify the Reactor Coolant Dose Equivalent I-131 Specific Activity Limit
- Revise Containment Spray Additive (NaOH) tank limits
- Revise the definition of Dose Equivalent I-131 to include reference to ICRP-30

The amendment also proposes a change in dose calculation methodology to the Alternate Source Term per 10CFR 50.67, Accident Source Term. Since Radiation Doses for Equipment Qualification are not addressed, this should be considered a partial AST Submittal.

2.0 PROPOSED CHANGE

Rochester Gas and Electric Corporation (RG&E) intends to modify the CREATS per Ginna Plant Change Record (PCR) 2000-0024, to improve system reliability, performance and redundancy. To provide a basis for explanation, a drawing of the new configuration is included as Figure 1. Upon completion of the modification, the present (normal) system will remain in place and will serve as normal HVAC, except that the existing Control Room Emergency Fan and Filter Unit (AKF07) will be either removed or abandoned in place. While in the accident mode, the normal HVAC system will be isolated by redundant leak tight dampers. The new CREATS will start and re-circulate the control room environment through High Efficiency Particulate Absorbers (HEPA) and Charcoal Filters to provide a safe environment for the Operators. The new system will have redundant trains and will be powered from safeguard power sources with Diesel Generator backup. Each train will provide a nominal 6000 CFM of re-circulation flow. A new actuation signal will be installed to automatically shift the CREATS to the emergency mode upon receipt of a Safety Injection signal in addition to the already existing high radiation and toxic gas initiation signals. A detailed description of the plant modification can be found in the Design Criteria (Attachment 7) and the draft UFSAR section 6.4 changes (Attachment 8).

Specifically, Ginna Station Technical Specifications are to be revised as follows (see attachment 3 for markups):

- Section 1.1 - The Dose Equivalent I-131 will be changed to reflect the thyroid dose conversion factors consistent with the new analysis.

- Section 3.3.6 - The CREATS instrumentation section will be changed to identify the new Safety Injection CREATS actuation signal as assumed in the new dose analysis and to eliminate CORE ALTERATIONS from the Mode of Applicability. This section is also being revised by a separate submittal (reference 9), and following approval of that submittal the retyped version of this section will be submitted along with a copy of the marked up bases sections.

- Section 3.4.16 - RCS Specific Activity Limits will be revised to eliminate Figure 3.4.16-1 and provide a single limit for Dose Equivalent I-131 specific activity consistent with the new dose analysis.

- Section 3.6.6 - This section will be revised to eliminate the requirement and associated surveillance for Containment Post-Accident Charcoal Filters. The new analysis was performed without crediting these filters with satisfactory dose consequences. The revised section will remain consistent with Westinghouse Standard Technical Specifications, NUREG-1431 (reference 7), with respect to containment cooling, iodine removal, and ph control.

  SR 3.6.6.8 will be revised to reflect the actual amount of NaOH required to meet the ph requirement consistent with the new analysis. The present limit is overly restrictive. The proposed limit is conservative in that it is approximately three times the volume required by the Ginna UFSAR Section 6.1.2.1.4 and associated reference calculations, which is in itself conservative with respect to sump ph requirement.

  SR 3.6.6.9 will be revised to add an upper limit of NaOH concentration consistent with Ginna Station Safety Evaluation SEV-1057, Revision 2 (reference 12).

- Section 3.7.9 - The CREATS section will be changed to reflect the new system configuration. The change reflects Westinghouse Standard Technical Specifications (NUREG-1431, Rev 2) and Tech Spec Task Force (TSTF) Traveler TSTF-448, as modified for Ginna specific considerations.

- Section 5.5.10 - The Ventilation Filter Testing Program (VFTP) will be changed to reflect the removal of the Containment Post-Accident Charcoal Filter and the revised testing requirements for the new CREATS.

- Section 5.5.16 - Add the requirement for a Control Room Integrity Program as prescribed in TSTF-448.

- Section 5.6.7 - Add a Control Room Emergency Filtration System Report as prescribed in TSTF-448.

As part of this modification, RG&E has elected to recalculate the Control Room and site boundary doses using the Alternate Source Term Methodology per NRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (reference 3). This is a methodology change from previously approved Ginna Station evaluations. New x/Q values for the Control Room were calculated using the ARCON 96 computer code, supplemented by the guidance in Draft Regulatory Guide DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (reference 6). New off-site x/Q values were calculated (used in the Locked Rotor Dose Analysis only) using the PAVAN code and guidance in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1 (Reference 14). The results of these analyses support the proposed change to eliminate the requirement for the containment post accident charcoal filter trains in Technical Specification LCO 3.6.6 and substitute a single limit of 60uCi/gm Dose Equivalent I-131 in place of the curve in Figure 3.4.16.1. The analysis also verifies that the design of the new CREATS will maintain the Operators dose below allowable limits, assuring that they will be capable of performing their required duties during a DBA.

The present configuration of the Containment Post Accident Charcoal Filter System requires a damper shift and redirection of airflow in the event of an accident. That is, the normal discharge flow path for Containment Recirculation Fan Coolers (CRFCs) A and C are isolated and redirected to the charcoal filters. The appropriate surveillance requirements are maintained to ensure the continued operability of the CRFC A and C and the associated HEPA filters.

In summary, the proposed amendment will reflect the post modification CREATS system configuration and other related changes. In addition, the requirement for Containment Post Accident Charcoal filters will be removed and the Reactor Coolant I-131 dose equivalent limit will be adjusted as supported in the updated dose analysis.

3.0 BACKGROUND

RG&E has undertaken a voluntary initiative to upgrade the Control Room Emergency Air Treatment System by installing a new system including fans, filters, dampers and environmental controls. The new system will improve reliability, redundancy and environmental conditions for the operators during accident conditions.

To minimize the impact upon plant operation and scheduling, the new equipment will be installed and tested during plant operation, with the exception of certain electrical ties. This will be performed without breaching the Control Room boundary as follows:

Short ducts sealed with blind flanges will be installed on the inside of the Control Room walls to provide a connection point for the interior ductwork (see A on Figure 1). Holes will then be cut from the outside of the control room wall along the internal radius of the short ducts. The interior blind flange plates will stay in place until the external system is installed and pressure tested.

The new fans, filters and other components will be assembled and connected to the short ducts at the holes described above. The new system external to the control room will be constructed to leak tight standards, and will be pressure tested for leak tightness. After testing, the flanges will be removed and the interior ductwork connected. This new equipment may then be operated as necessary for heating and cooling purposes, with the added benefit of providing filtration in the event of an accident.

After the new equipment is operational, new dampers AKD21, AKD22, AKD23 and AKD24 will be installed to provide emergency mode isolation of the normal HVAC system. These 4 dampers will be located in the Control Room/Relay Room stairwell. RG&E Design Analysis DA-NS-2000-070, Control Room Dose Simulation Removal of MUX Room Temporary Stairwell Enclosure (reference 8) documents the acceptability of the stairwell as an extended Control Room boundary.

Prior to opening the ductwork for installation of the new dampers (inside the stairwell), the existing CREATS equipment will be isolated and declared inoperable. This will require entry into current Tech Spec 3.7.9, Condition A. It is anticipated the plant will be in this LCO for less than 30 days. The new CREATS will be operated during this period for temperature control and will provide a contingency for filtration and activity removal in the event of an accident.

After the four new dampers are installed the system will be operationally tested and turned over to Operations. The new proposed Technical Specifications will then be implemented and a tracer gas in-leakage test performed to verify the in-leakage assumptions used in the dose analysis. Attachments 7, 8, and 9 provide additional details concerning the new CREATS.

4.0 TECHNICAL ANALYSIS

As part of this submittal, RG&E performed a detailed analysis of the radiological dose and toxic gas consequences of the proposed modification. Each of the Design Basis Accidents in UFSAR Chapter 15 that had a calculation for dose consequences was recalculated using Alternate Source Terms per 10CFR50.67 (reference 2), and expressed in TEDE for comparison with the limits contained in Regulatory Guide 1.183 (reference 3). Additionally, to standardize and maintain consistency with current guidance, RG&E performed the remaining Analyses listed in Regulatory Guide 1.183 with the exception of the Dose for Equipment Qualification. The toxic gas analysis (Attachment 2 and reference 13) is consistent with Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Revision 1 (reference 10). The following specific analyses were performed:

Dose Analysis

- Atmospheric Dispersion, x/Q
- Iodine Spiking
- Loss of Coolant
- Fuel Handling Accident
- Main Steam Line Break
- Steam Generator Tube Rupture
- Locked RCP Rotor
- Rod Ejection Accident
- Tornado Missile in Spent Fuel Pool
- Waste Gas Decay Tank Rupture

Other Analysis

- Toxic Gas Effects

The above analyses demonstrate that the changes are acceptable from a dose and chemical perspective. The specific results of the analyses are included in Attachment 1, Alternative Source Term and Control Room Emergency Ventilation System Submittal.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed amendment would revise the operating license as follows:

- Reflect the new Control Room Emergency Air Treatment System (CREATS)
- Change the dose calculation methodology to the Alternate Source Term (AST) per 10CFR 50.67, Accident Source Term
- Implement new industry guidance related to CREATS
- Add the requirement for a Control Room Integrity Program
- Add reporting requirements for an inoperable boundary
- Eliminate the requirement for Containment Post Accident Charcoal Filters
- Modify the Reactor Coolant Dose Equivalent I-131 Specific Activity Limit
- Revise Containment Spray Additive (NaOH) tank limits
- Revise the definition of Dose Equivalent I-131 to include reference to ICRP-30

RG&E has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10CFR50.92, "Issuance of Amendment," as discussed below.

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

Response: No.

The function of the CREATS is to provide a safe environment for the operators in the event of an accident, and thereby allow them to perform their accident mitigation responsibilities. The physical changes to the CREATS were designed to enhance the ability of the system to perform that function. The new system is an improvement in reliability, redundancy and leak tightness over the existing system. The change in design has no impact on accident initiation frequencies. Therefore the physical changes to the plant do not increase the probability or consequences of a previously evaluated accident.

The proposed Technical Specification changes involving the CREATS reflect the new system configuration and current industry guidance. The specifications ensure system functionality and protection of the operators under postulated accident conditions.

The new dose analysis indicates that the radiation dose to the operators and the public is acceptable without crediting the post accident charcoal filters removed from Technical Specification 3.6.6 and 5.5.10, and also bounds the change to the Reactor Coolant System activity limits in Technical Specification 3.4.16. The change to the dose conversion factor definition in Technical Specification section 1.1 is consistent with the new analysis.

The reference to ICRP-30 in the Dose Equivalent I-131 definition is consistent with the new analysis and Standard Tech Specs, NUREG1431.

All calculated doses are within the regulatory limits prescribed in 10CFR50.67. In addition, with the exception of one calculated Exclusion Area Boundary (EAB) dose, all dose numbers are within the guidelines of Reg Guide 1.183 and Standard Review Plan (SRP) 15.0.1. This above-mentioned dose is in one particular direction from the source. The associated accident is the Locked Rotor Accident, which was not previously evaluated for dose at Ginna. The 100% fuel failure assumption used in this accident is widely considered to be overly conservative. Additionally, extra margin is built into the calculation because RG&E assumed 500 gallons per day (GPD) of Steam Generator (SG) tube leakage per SG. Since the primary release pathway for this accident is SG tube leakage, and Reg Guide 1.183 (reference 3) allows an assumed tube leakage equal to the Tech Spec allowable leakage (~150 GPD/SG at Ginna), RG&E assumed a release rate of ~3.3 times greater than required. The calculated dose (2.7 Rem) is well below the regulatory limit of 25 Rem and only slightly greater than the published guideline of 2.5 Rem. Given the localized nature, associated probability/risk, and conservatism in this analysis, the calculated dose is considered acceptable.

Iodine removal was not credited in the existing analysis of doses for Equipment Qualification. Therefore, even though the Containment Post Accident Charcoal Filters will be removed from Tech Specs as a result of this amendment, it is not necessary to re-analyze these doses.

The Toxic Gas in-leakage analysis is bounded by the assumed in-leakage in the dose analysis. The amendment also does not hinder or change the ability to mitigate smoke infiltration as described in NEI 99-03, Control Room Habitability Guidance.

This change has no impact on accident initiators, will not affect the ability of the operators to perform their designated functions, and removal of the requirement for CNMT Post Accident Charcoal Filters is shown to be acceptable. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

For the proposed changes, a different kind of accident would involve a situation where the operators would become incapacitated or otherwise be prevented from fulfilling their function. The new system differs in that the cooling in the emergency mode is from direct expansion of R-22 refrigerant. A rupture of the coils could introduce the refrigerant into the Control Room environment. However, the charge of refrigerant R-22 in cooling system will be limited such that a rupture in the cooling coils would not exceed nationally accepted toxicity standards.

The radiation and/or toxic gas exposures are shown to be acceptable, and the ability of the plant to mitigate smoke infiltration has not changed. The new system will improve the environmental conditions in most situations and actually enhance the ability of the operators to perform their functions.

Given the above, an event that would result in preventing the operators from fulfilling their safety functions is not introduced by this change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The new analysis was performed without crediting the existing Containment Post Accident Charcoal Filters and indicated that the Control Room and off-site doses remain within the required limits. Removal of the Post Accident Charcoal Filters from Technical Specification will not impact the operators' ability to function or significantly increase dose to the public.

The new Technical Specification surveillance limits for NaOH tank level and concentration establish criteria acceptable to meet the assumptions in the dose analysis.

The changes to the VFTP program in Technical Specification reflect the removal of the Containment Post Accident Charcoal Filters consistent with the analysis, and the surveillance limits consistent with the new CREATS design.

The use of AST represents a change to a standardized and accepted dose calculation method.

The function of the CREATS system is to protect the operators and allow them to perform the necessary accident mitigation tasks. The proposed changes to the CREATS enhance this ability through improved redundancy and system operation. The analysis demonstrates that the Control Room will remain within prescribed limits during the design basis accidents. The operators will be able to perform their function and the public will be protected.

Therefore, the proposed change does not involve a significant reduction in a margin to safety.

Based on the above, RG&E concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10CFR50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements

10CFR50 Appendix A General Design Criteria (GDC)19 (reference 1)

GDC 19 states in part, "A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident." GDC 19 also provides allowance for compliance under 10CFR50.67 provision for Alternate Source Term.

The new CREATS system will have two redundant 100% capacity trains and Limiting Conditions for Operation which have been approved as part of the Westinghouse Owners Group (WOG) Standard Technical Specifications. As such, the proposed changes ensure the above stated criteria are met during all modes of applicability.

10CFR50.67 Accident Source Term

This part has provisions for compliance using the alternate source term. The technical summary of analysis, Section 4, demonstrates compliance with this part and GDC 19. The change in methodology to the alternate source term is in compliance with this part and Reg Guide 1.183.

NUREG-0800, Standard Review Plan, section 6.4 (reference 4), was reviewed and the proposed changes are shown to be consistent with this guidance (Attachment 9).

6.0 ENVIRONMENTAL CONSIDERATION

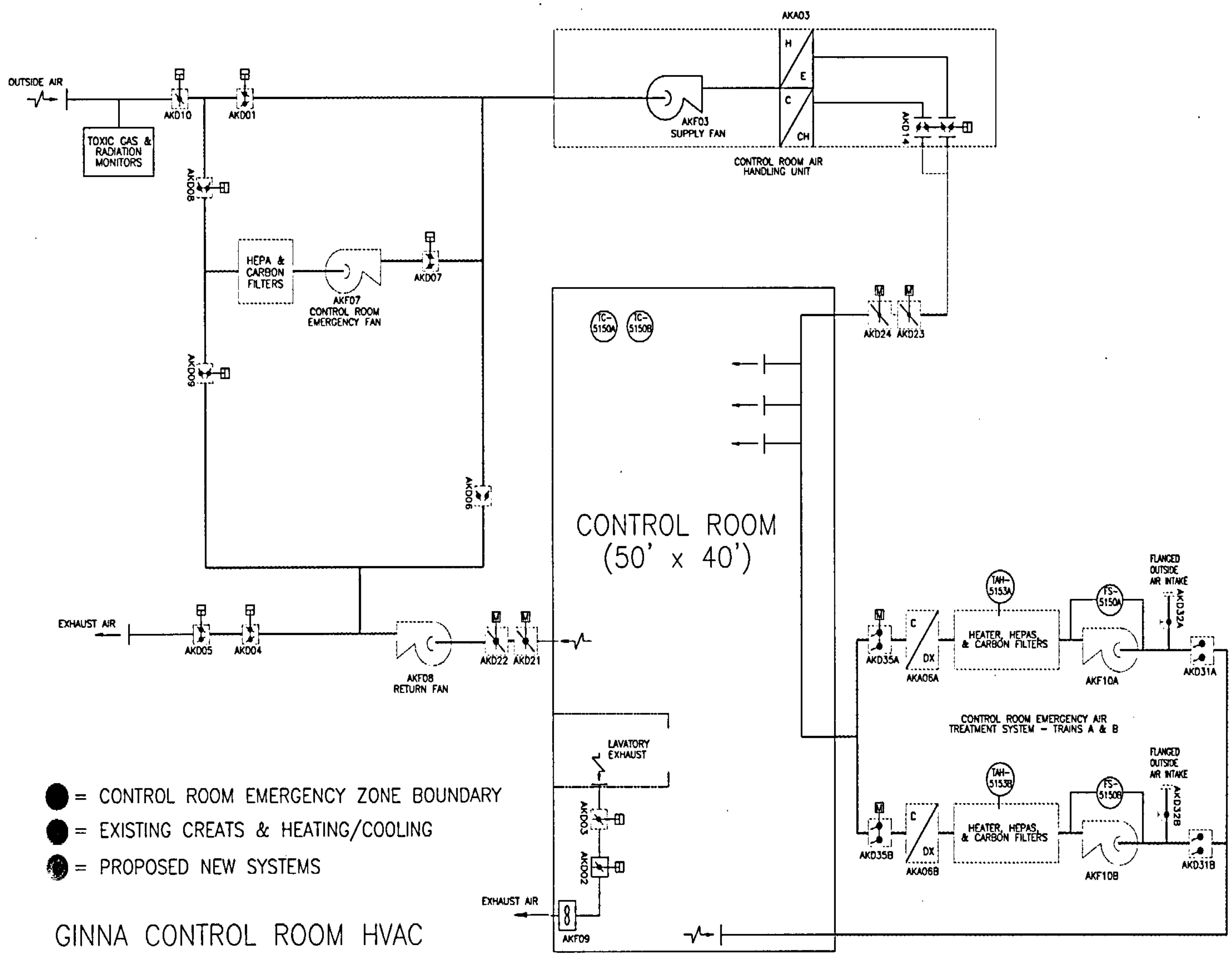
A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment (reference 5).

7.0 REFERENCES

1. 10CFR50 Appendix A, Criterion 19 (GDC 19), "Control Room"
2. 10CFR50.67, "Accident Source Term"
3. Reg Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".
4. NUREG 0800, Standard Review Plan, Section 6.4, "Control Room Habitability System".
5. 10CFR51.22, "Criterion for Categorical Exclusion; Identification of Licensing and Regulatory Actions Eligible for Categorical Exclusion or Otherwise Not Requiring Environmental Review".
6. DG-1111, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants"
7. NUREG-1431, Revision 2, "Standard Technical Specifications Westinghouse Plants".
8. DA-NS-2000-070, "Control Room Dose Simulation Removal of MUX Room Temporary Stairwell Enclosure".
9. Letter from Robert C. Mecredy (RG&E) to Guy S. Vissing (NRC), "Application for Amendment to Facility Operating License Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation Change (LCO 3.3.6)", dated May 3, 2001.

10. Regulatory Guide 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release, Revision 1.
11. NEI 99-03, Control Room Habitability Guidance
12. Ginna Station Safety Evaluation SEV-1057, 18 Month Fuel Cycle, Revision 2.
13. Ginna Station Design Analysis DA-NS-2000-053, Control Room Toxic Hazards Analysis
14. Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Revision 1
15. Ginna UFSAR, Revision 17
16. NUREG 0800, Standard Review Plan (SRP), Section 15.0.1, July 2000

Figure 1



OUTSIDE AIR
TOXIC GAS & RADIATION MONITORS
AKD10
AKD01
AKD08
HEPA & CARBON FILTERS
AKF07 CONTROL ROOM EMERGENCY FAN
AKD07
AKD09
AKD06
AKA03
AKF03 SUPPLY FAN
CONTROL ROOM AIR HANDLING UNIT
AKD14
AKD24
AKD23
CONTROL ROOM (50' x 40')
EXHAUST AIR
AKD05
AKD04
AKF08 RETURN FAN
AKD22
AKD21
LAVATORY EXHAUST
AKD03
AKD02
EXHAUST AIR
AKF09
HEATER, HEPAS, & CARBON FILTERS
AKD35A
AKA06A
AKF10A
AKD31A
AKD32A
FLANGED OUTSIDE AIR INTAKE
CONTROL ROOM EMERGENCY AIR TREATMENT SYSTEM – TRAINS A & B
AKD35B
AKA06B
AKF10B
AKD31B
AKD32B
= CONTROL ROOM EMERGENCY ZONE BOUNDARY
= EXISTING CREATS & HEATING/COOLING
= PROPOSED NEW SYSTEMS
GINNA CONTROL ROOM HVAC

Attachment 1

Alternative Source Term and Control Room Emergency Ventilation System Summary of Radiological Analysis for UFSAR Chapter 15 Analysis

Rochester Gas and Electric Corporation
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R. E. Ginna Station

Docket Number 50-244

Summary of Radiological Analyses

Alternative Source Term and Control Room Emergency Ventilation System Submittal

May 2003

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1.0 Summary of Radiological Analysis

Each of the below accidents was analyzed for dose consequences using the Alternative Source Term Methodology per Regulatory Guide 1.183. All dose results are expressed in terms of TEDE for comparison with the appropriate limits. The accident consequences were calculated for both the Control Room Operator and the public at the Exclusion Area Boundary (EAB) and the Low Population Zone (LPZ). The following table summarizes the results of the analysis.

TABLE 1.1
ALTERNATE SOURCE TERM DOSE ANALYSIS SUMMARY

Accident	EAB Max. 2-hour		LPZ		Control Room	
	Limit	Dose	Limit	Dose	Limit	Dose
LOCA	25.0	5.92	25.0	1.06	5.0	3.03
FHA - CNMT	6.3	1.1	6.3	0.07	5.0	1.2
FHA-AUX	6.3	0.31	6.3	0.02	5.0	0.09
MSLB[1]	2.5	1.05	2.5	0.15	5.0	0.64
MSLB[2]	25.0	0.15	25.0	0.03	5.0	0.18
SGTR[1]	2.5	0.22	2.5	0.02	5.0	0.14
SGTR[2]	25.0	0.71	25.0	0.05	5.0	0.88
Locked Rotor	2.5	2.75[4]	2.5	0.55[4]	5.0	3.72
Rod Ejection	6.3	1.47	6.3	0.24	5.0	1.04
SFP- TMA	6.3	0.07[3]	-	-	5.0	0.06
GDT Rupture	0.5	0.28	0.5	0.02	5.0	0.07/0.10

[1] Accident Initiated Iodine Spike
[2] Pre-Accident Iodine Spike
[3] EAB X/Q = 1.74E-6 calculated as discussed in Section 2.7
[4] EAB and LPZ X/Q calculated by PAVAN, see Section 2.8

2.0 Atmospheric Dispersion (X/Q)

The atmospheric dispersion factors currently described within the UFSAR were reviewed as part of the control room ventilation system upgrade. As a result of this review, the atmospheric dispersion factors for the control room intake were recalculated as described below. The atmospheric dispersion factors for the EAB and LPZ are described in Section 2.8.

The atmospheric dispersion factors for each pathway from on-site source to control room intake were recalculated using the ARCON96 code (Reference 1) combined with the draft 2 Reg. Guide DG-1111 methodology (Reference 2).

The meteorological data collected by a Regulatory Guide 1.23 system for the years 1992, 1993, and 1994 was used in the calculations. This data is considered to be typical of any time period. This data was readily available and used in prior submittals. The data covered 26,304 hours, of which 512 hours were missing or invalid. This represents approximately 2% which is within the ARCON96's default setting of 10%.

The wind speed statistics for a typical year (1992) are:

Average wind speed:	4.16 m/sec
Maximum:	24.5 m/sec
Total hours (including invalid):	8572
Invalid hours:	212

The stability distribution for the same year (1992) was:

Stability Class	Duration (hr)
A	739
B	466
C	351
D	3132
E	2389
F	835
G	660

2.1 Containment Leakage

The containment shell is modeled as a diffuse vertical area source. This source is used in the dose calculations for LBLOCA, and the containment leakage portion of a control rod ejection accident. The source width is the containment O.D. and the source height is the distance from ground to the top of the containment dome. This is consistent with Reference 2, Figure 1. The diffuse source model is used because leakage is assumed to be distributed over the containment surface and all penetrations, not isolated to a specific point.

The source to receptor distance uses the shortest horizontal distance from the containment surface to the intake and assumes the source and receptor are at the same height. This results in the shortest source to receptor distance as illustrated by points C and B on Figure 2.1.B. The ARCON96 input parameters and resulting X/Qs are presented on Table 2.1 and Figures 2.1A and 2.1B.

TABLE 2.1 CONTAINMENT LEAKAGE INPUT AND RESULTS	
Distance to receptor, m	32
Intake height, m	13.8
Direction to source, degrees	247
Release type	ground level, diffuse vertical area
Release height, m	13.8
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m σ_{y0} σ_{z0}	 5.7 5.9
Lower measurement height, m	10
Upper measurement height, m	100
Elevation difference, m	0

TABLE 2.1 CONTAINMENT LEAKAGE INPUT AND RESULTS	
Resulting X/Q, sec/m^3	
0-2 hr	1.57 E-03
2-8 hr	1.12 E-03
8-24 hr	4.47 E-04
1-4 days	3.69 E-04
4-30 days	3.10 E-04

FIGURE 2.1A - CONTAINMENT AND CONTROL BUILDING ELEVATIONS

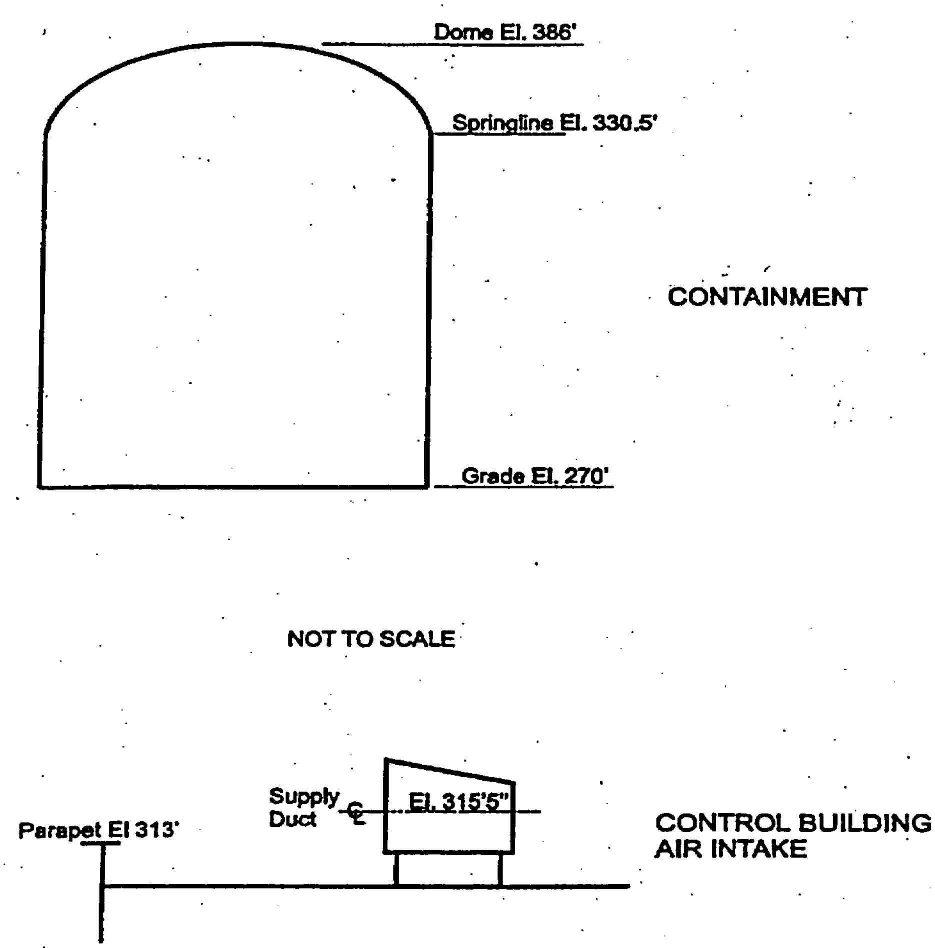
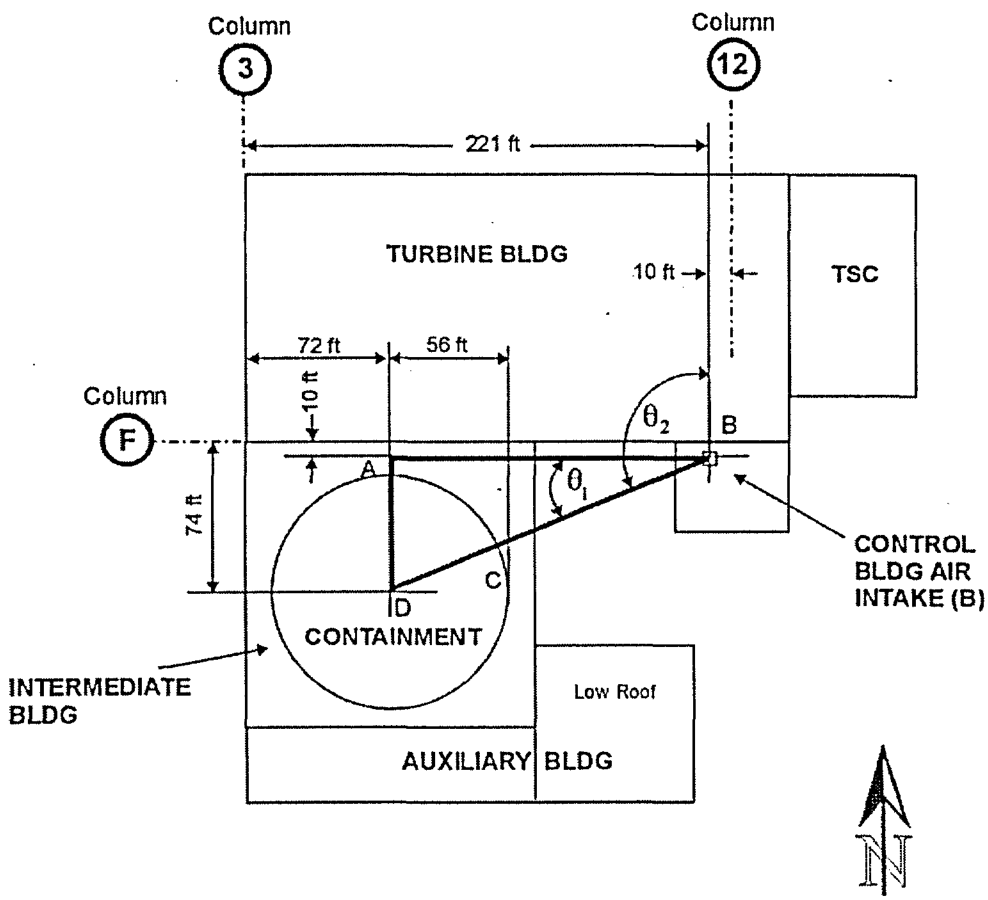


FIGURE 2.1B - CONTAINMENT LEAKAGE PLAN VIEW



2.2 Containment Equipment Hatch (Roll-Up Door)

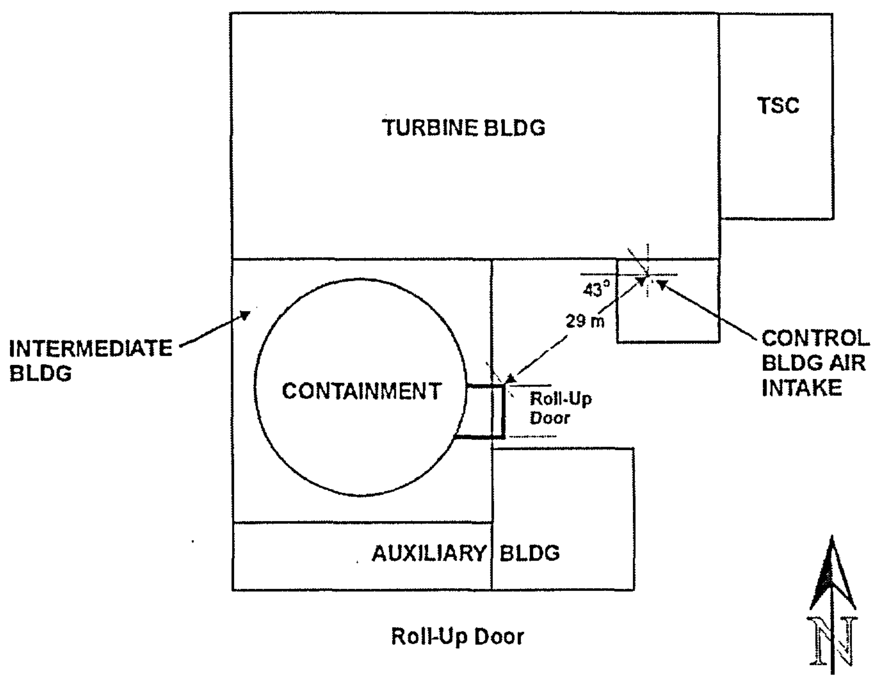
This vertical area source is used for the fuel handling accident in Containment. In this case, all Containment leakage is assumed to come from the equipment hatch, a large penetration located in the south-east sector of the Containment perimeter. During refueling, the hatch is removed, and the open penetration is covered by a roll-up door. The source dimensions are based on face area of the roll-up door. Radioactivity is postulated to leak through the open hatch, to the environment, through the perimeter seals of the roll-up door. The calculation uses:

1. The shortest horizontal distance between the door perimeter and the Control Room Intake
2. A diffuse vertical source is assumed. The dimensions being that of the roll-up door (23'6" wide, 22' high).
3. The assumed release height is equal to the distance from grade to the top of the roll-up door. This results in the shortest source - receptor distance.
4. The cross-section area of Containment was assumed for the building wake effect (1071 m^2). A sensitivity run was made where the wake area was doubled. There was an insignificant change in X/Q. The ARCON96 input parameters and resulting X /Q are presented on Table 2.2 and Figure 2.2.

TABLE 2.2 CONTAINMENT EQUIPMENT HATCH INPUT AND RESULTS	
Distance to receptor, m	29
Intake height, m	13.8
Direction to source, degrees	227
Release type	ground level, diffuse vertical area
Release height, m	6.7
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m σ_{y0} σ_{z0}	 1.2 1.1
Lower measurement height, m	10

TABLE 2.2 CONTAINMENT EQUIPMENT HATCH INPUT AND RESULTS	
Upper measurement height, m	100
Elevation difference, m	0
Resulting X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-30 days	 5.64 E-03 4.69 E-03 1.66 E-03 1.58 E-03 1.31 E-03

FIGURE 2.2 - ROLL-UP DOOR PLAN VIEW



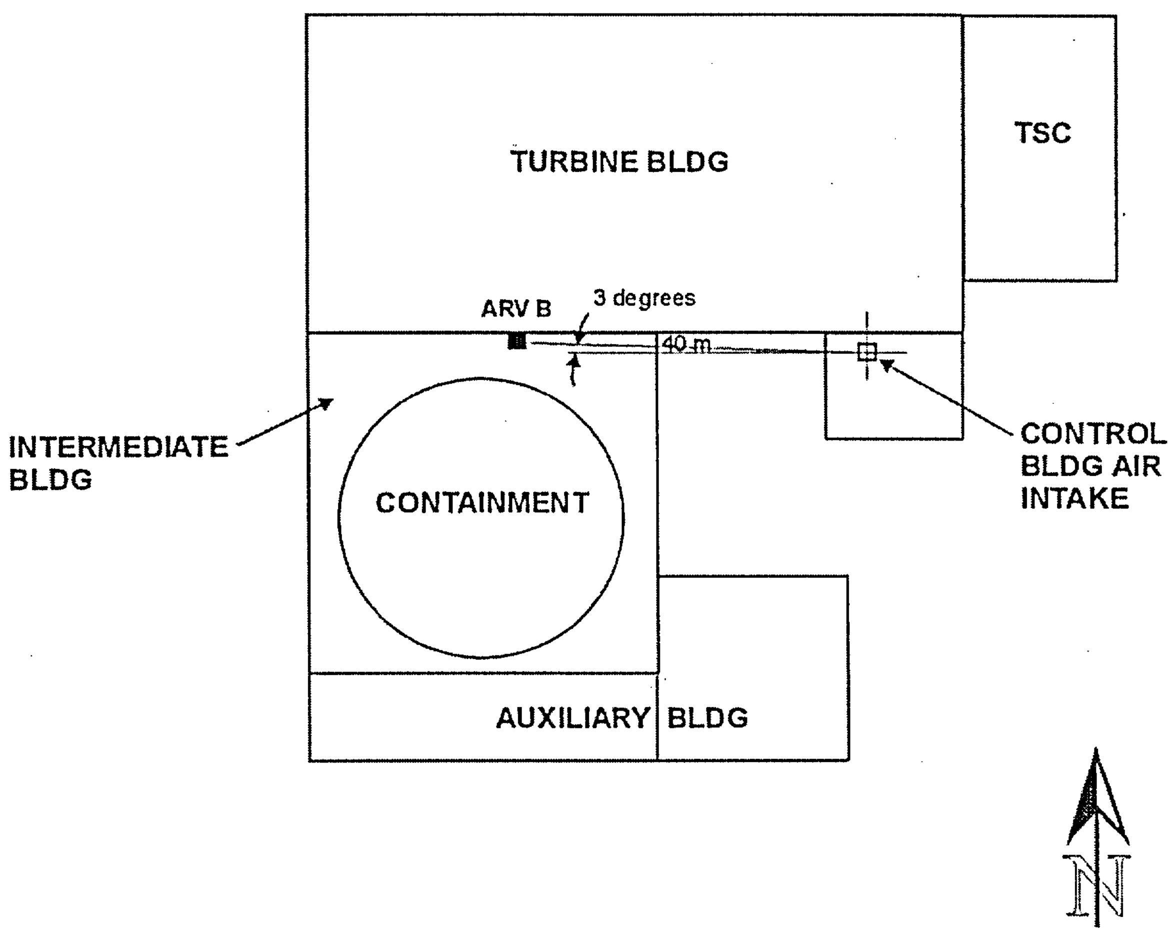
Roll-Up Door

2.3 Atmospheric Relief Valves (ARVs)

This source is used for releases from the steam generators. The pathway is based on the ARV discharge that is closest to the Control Room Intake. This will result in larger X/Qs. The discharge of the ARV is modeled as a ground-level point source rather than an elevated vent since Reference 2 advises against using the vent release model, pending further NRC evaluation. The assumed release height is equal to the distance from grade to the vent point. The cross-section area of Containment was assumed for the wake area (1071 m^2). A sensitivity run was made where the wake area was doubled. There was an insignificant change in X/Q. The ARCON 96 input parameters and resulting X/Q are presented on Table 2.3 and Figure 2.3.

TABLE 2.3 ATMOSPHERIC RELIEF VALVES INPUT AND RESULTS	
Distance to receptor, m	40
Intake height, m	13.8
Direction to source, degrees	273
Release type	ground level, point
Release height, m	22
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m σ_{y0} σ_{z0}	 0 0
Resulting X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr 1-4 days 4-30 days	 3.66E-03 2.49E-03 1.07E-03 7.86E-04 7.17E-04

FIGURE 2.3 - ARV GROUP B PLAN VIEW



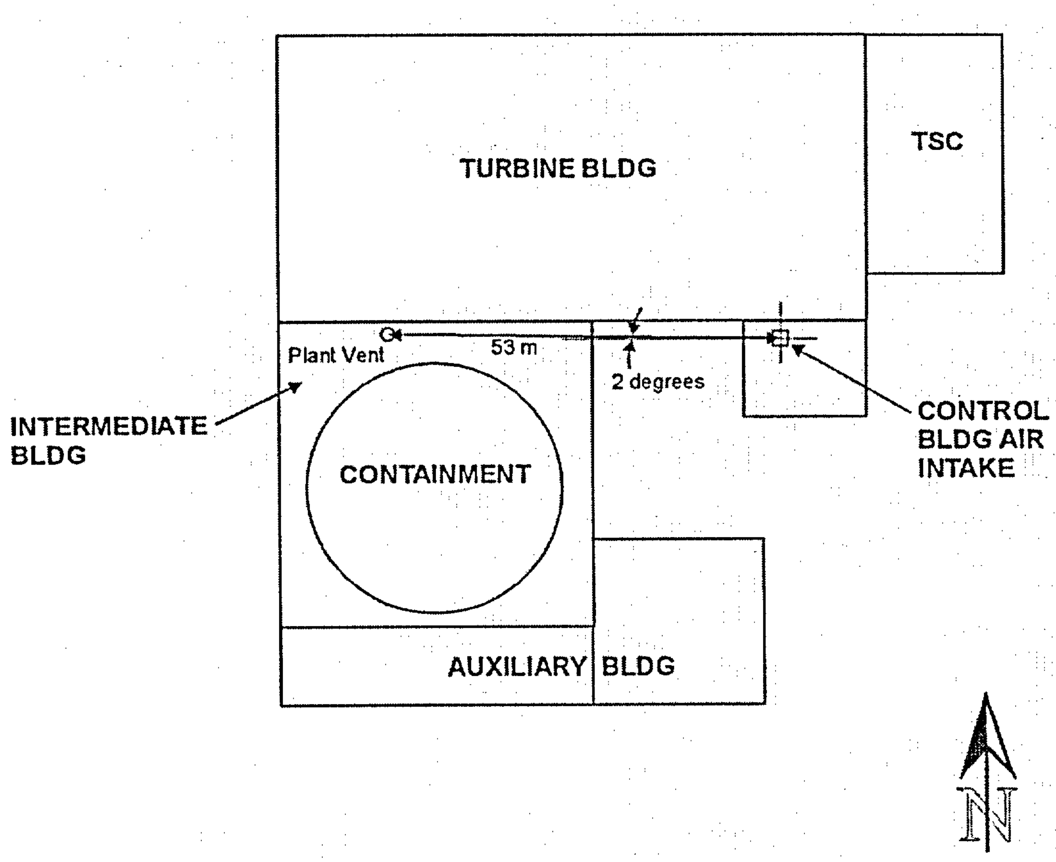
2.4 Plant Vent

This source is used for releases from a fuel handling accident in the spent fuel pool. The vent is modeled as a horizontal area source, rather than a vent source, based on guidance in Reference 2, which advises against using the vent release model pending further NRC evaluation. The assumption of an area source is considerably more conservative than the vent source assumption and only slightly less conservative than a point source. For the horizontal area source, the horizontal diffusion coefficient is based on the vent diameter (55") and the vertical coefficient is set to zero. The assumed release height is equal to the distance from grade to the vent point. The wake area of 1071 m^2 is again assumed. Doubling this value had an insignificant affect on X/Q. The ARCON96 input parameters and resulting X/Q are presented on Table 2.4 and Figure 2.4.

TABLE 2.4
PLANT VENT INPUT AND RESULTS

Distance to receptor, m	53
Intake height, m	13.8
Direction to source, degrees	272
Release type	ground level, diffuse horizontal area
Release height, m	36
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m	
σ_{y0}	0.23
σ_{z0}	0
Resulting X/Q, sec/m^3	
0-2 hr	1.79E-03
2-8 hr	1.15E-03
8-24 hr	4.95E-04
1-4 days	3.71E-04
4-30 days	3.29E-04

FIGURE 2.4 - PLANT VENT PLAN VIEW

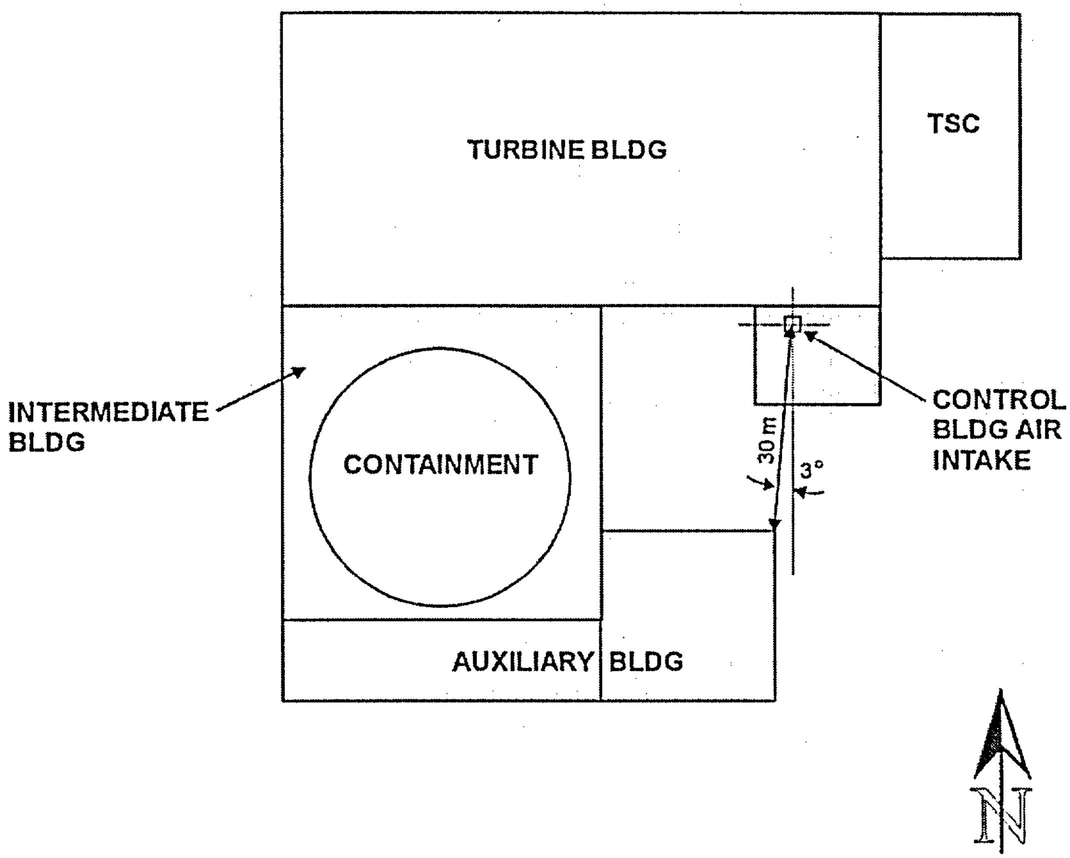


2.5 Auxiliary Building Leakage

This source is used when ECCS leakage is considered. The subgrade floors of this building contain ECCS equipment that is postulated to leak. The source is modeled as a vertical area source assumed to be the building wall closest to the Control Room Intake. The building north wall is modeled which approximates the cross-sectional area perpendicular to the line of site from the building surface to the control room intake. The assumed release height is the distance from grade to the top of the Auxiliary Building. The wake area equivalent to the Containment cross-section area is again assumed. The shortest source - receptor distance is calculated from the corner of the Auxiliary Building to the Control Room Intake. The ARCON96 input parameters and resulting X/Q are presented on Table 2.5 and Figure 2.5.

TABLE 2.5 AUXILIARY BUILDING LEAKAGE INPUT AND RESULTS	
Distance to receptor, m	30
Intake height, m	13.8
Direction to source, degrees	183
Release type	ground level, diffuse vertical area
Release height, m	13
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m	
σ_{y0}	3.9
σ_{z0}	2.1
Resulting X/Q, sec/m^3	
0-2 hr	3.89E-03
2-8 hr	2.99E-03
8-24 hr	9.63E-04
1-4 days	8.98E-04
4-30 days	8.23E-04

FIGURE 2.5 - AUXILIARY BUILDING LEAKAGE PLAN VIEW

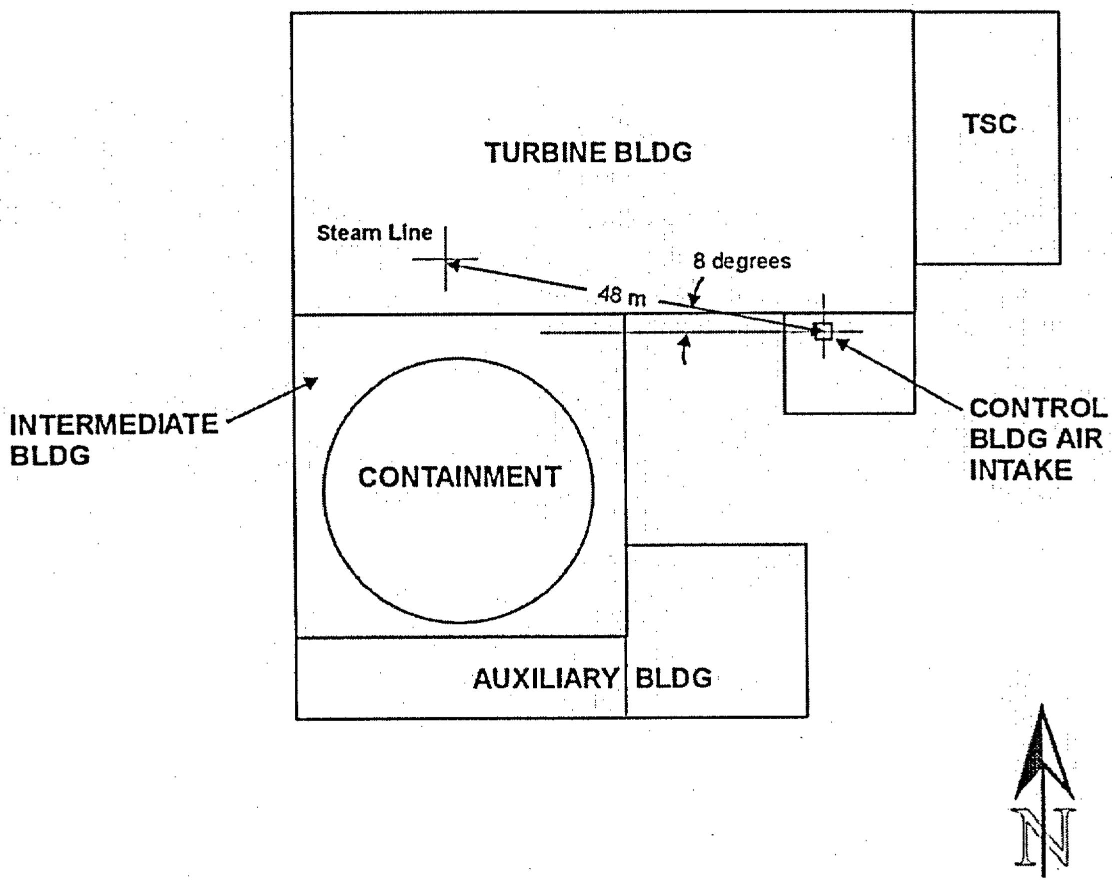


2.6 Main Steam Header Turbine Building

This source is used to model activity released from a main steamline break. The rupture site is assumed to be inside the Turbine Building on the mezzanine level. (See Section 7.1 for additional details.) Since the released steam is assumed to blow-out windows and metal siding of the Turbine Building, no confinement of the plume is assumed. The source is modeled as a ground level point source. The distance to the receptor is that from the header midpoint to the Control Room Intake. The release height is the distance from grade to the top of the header. The wake area equivalent to the Containment cross-section area is again assumed. The ARCON96 input parameters and resulting X/Q are presented on Table 2.6 and Figure 2.6.

TABLE 2.6 MAIN STEAM HEADER TURBINE BUILDING INPUTS AND RESULTS	
Distance to receptor, m	48
Intake height, m	13.8
Direction to source, degrees	278
Release type	ground level, point source
Release height, m	4
Building area, m^2	1071
Sector width constant	4.3
Surface roughness	0.2
Initial diffusion coefficients, m	
σ_{y0}	0
σ_{z0}	0
Resulting X/Q, sec/m^3	
0-2 hr	2.57E-03
2-8 hr	1.92E-03
8-24 hr	8.08E-04
1-4 days	5.77E-04
4-30 days	5.50E-04

FIGURE 2.6 - STEAM LINE PLAN VIEW



2.7 Tornado Missile

The tornado missile accident assumes that a utility pole, propelled by the wind, penetrates the Auxiliary Building roof and impacts fuel stored in the spent fuel pool. The specific location of the impact cannot be predicted. Thus, the shortest source-receptor distance is conservatively assumed. The source is modeled as a ground level point source. The release height is the distance from grade to the spent fuel pool surface. The wake area is again assumed equivalent to the cross-section area of Containment.

The calculation of atmospheric dispersion for tornado conditions is a unique task that cannot be performed with ARCON96. The primary reason that ARCON96 can't be used is the lack of meteorological data for a tornado. Further, if data were available, the duration of a tornado is too short for ARCON96 to provide a meaningful average. An ARCON96 calculation typically averages 1 to 5 years of hourly meteorological data. A tornado would provide two data points at most.

While the use of the ARCON96 code is not practicable for tornado conditions, the use of the dispersion models executed by ARCON96 may be used to conservatively estimate dispersion. The CONHAB module of the HABIT code calculates a single, direction and time-independent dispersion value using the basis dispersion models developed for ARCON96.

CONHAB is used to calculate dispersion factors for tornado wind speeds and to also show the sensitivity of the model to stability class. The input and results for these cases is summarized in Table 2.7. The basis for the selection of wind speed and stability class is as follows:

- wind speed

  The range is 24.5 to 60 meters/sec. 24.5 meters/sec is the maximum recorded hourly wind speed during normal atmospheric conditions. 132 miles/hour (about 60 meters/sec) is the wind speed for the design basis tornado.

- Pasquill stability class

  Stability class is a user input to the CONHAB dispersion model. There are 7 stability classes, A through G. "A" represents extremely unstable conditions, and "G" represents stable conditions. Unstable conditions enhance dispersion. Stable conditions diminish dispersion. However, the dispersion model predicts a diminishing effect with increasing wind speed. Test cases are run to show this effect and also to show that there are no discontinuities or instabilities in the model due to increasing wind speed. The cases show converging X/Qs with increasing wind speed (Figure 2.7). For the cases listed on Table 2.7, Pasquill F provides some conservatism over Pasquill A, even at wind speeds up to 60 meters/sec. Therefore, the tornado X/Q will be based on Pasquill F and a wind speed of 24.5 meters/sec., since this combination results in a larger X/Q.

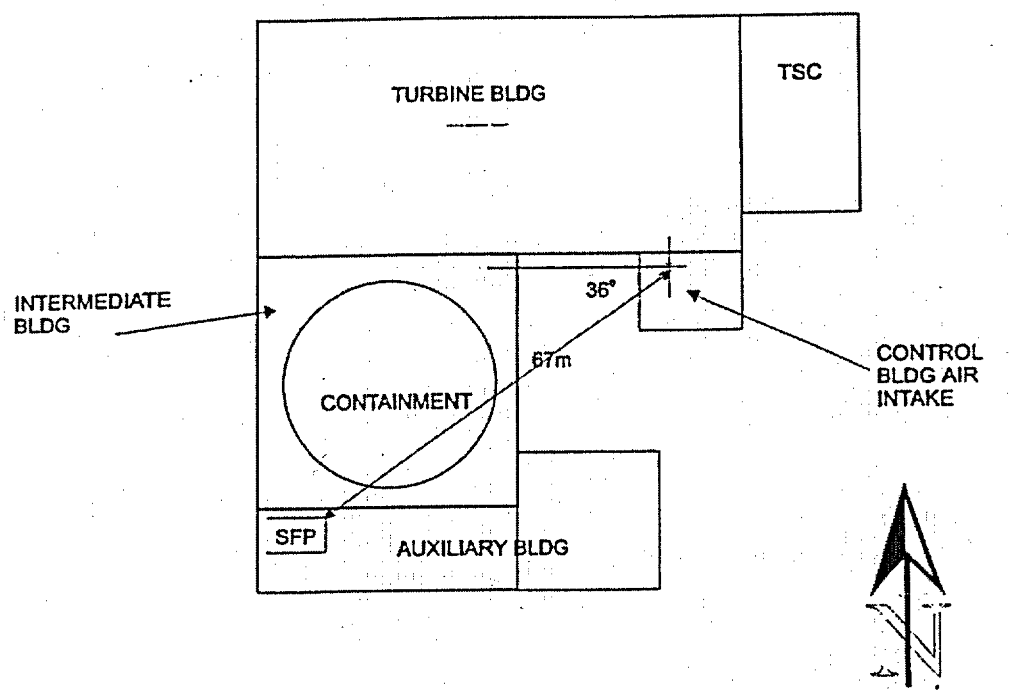
TABLE 2.7
CONHAB TORNADO MISSILE INPUT AND RESULTS

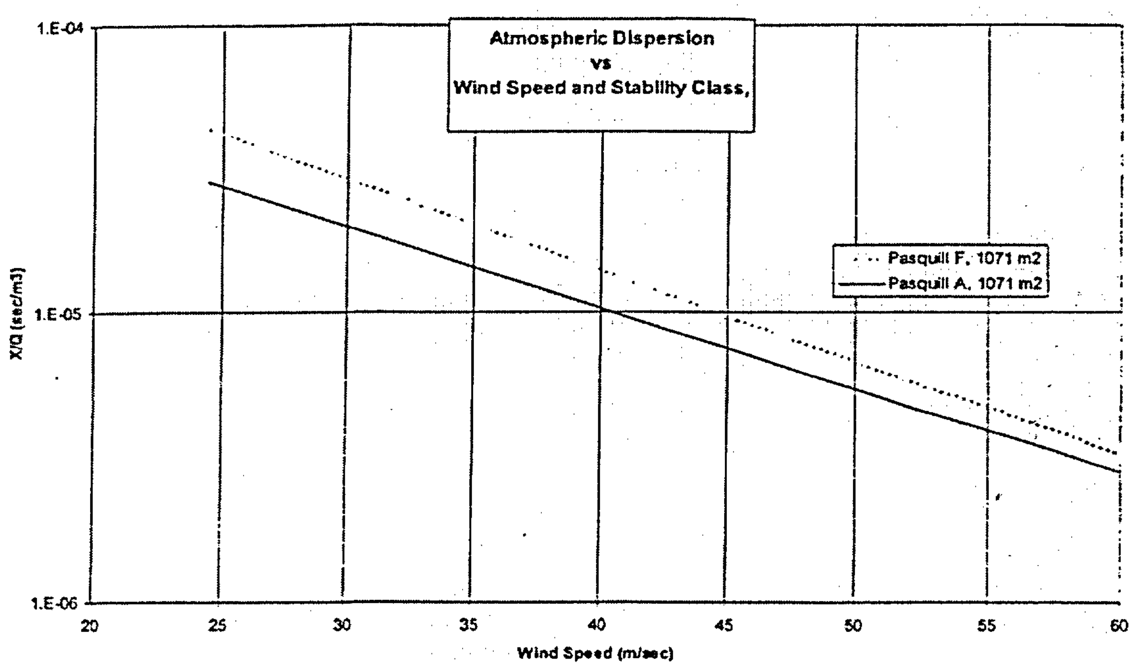
Parameter	CB1A	CB1F	CB2A	CB2F	CB3F
Distance to receptor, m	67				503
Intake height, m	13.8				
Release type	ground level, point source				
Release height, m	2.1				
Building area, m^2	1071				
Wind Speed, meters/sec	24.5	24.5	60	60	24.5
Stability Class	A	F	A	F	F
Resulting X/Q, sec/m^3	2.85E-5	4.36E-5	2.77E-6	3.03E-6	1.74E-6

Cases CB1A through CB2F are for the Control Room air intake.

Case CB3F is for the 503 meter EAB.

FIGURE 2.7 - SPENT FUEL POOL PLAN VIEW





2.8 EAB and LPZ Atmospheric Dispersion Factors

All the EAB and LPZ dose calculations used the current X/Qs presented in the Ginna UFSAR (Reference 3, Section 2.3.4.2.1) except for the tornado missile and locked rotor.

Due to the uniqueness of the tornado missile dose calculation and to maintain consistency between the control room calculation and the EAB calculation, the EAB calculation was done using the same methodology as used for the control room calculation. See Section 2.7 for a description of the methodology.

The ultra conservative assumptions used in the locked rotor dose calculation result in assuming 100% fuel failure. This assumption and the current UFSAR EAB X/Q results in an unrealistically high EAB dose. To obtain a more realistic dose, the EAB and LPZ X/Qs were recalculated using the PAVAN computer code. The recalculated values have only been used to calculate the locked rotor doses. The intent is to use these new values in any future calculations.

2.8.1 Current UFSAR Atmospheric Dilution Factors (Reference 3, Section 2.3.4.2.1)

Site boundary (0-2 hr)	4.8E-04
Low population zone	
0-8 hr	3.0E-05
8-24 hr	2.1E-05
1-4 days	8.6E-06
4-30 days	2.5E-06

2.8.2 Recalculated Atmospheric Dispersion Factors (PAVAN code)

The same meteorological data used to calculate the Control Room X/Q was used to recalculate the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) dispersion factors. The dispersion factors were calculated using KR PAVAN which is a PC-based version of the NRC's PAVAN code (Reference 9). Dispersion factors were calculated using the direction-dependent method and the direction-independent method. The direction-dependent method determined the 0.5 percent X/Q value for each of the 16 wind speed directions. The direction-independent method determined the 5 percent X/Q value for the overall site. The direction-dependent dispersion values were limiting, for both EAB and LPZ boundaries. The input assumptions are listed on Table 2.8.

The EAB dispersion factor calculated by the ENVLOP routine of PAVAN is a conservative bounding 0.5 percent X/Q value. Since the EAB dose can be limiting for certain accidents, such as locked rotor, a more accurate X/Q value is desired. Therefore, the output X/Q vs. percentile for the limiting sector is analyzed in a spreadsheet to obtain a more accurate value.

The results of the spreadsheet analysis are shown in Figures 2.8.1 and 2.8.2. A logarithmic trend line is fit to the data and the resulting 0.5% X/Q value (0-2 hours) is determined. The 0.5 percent code and spreadsheet values for the limiting sector (SW) are:

code value	3.32E-4
equation value using all data points	3.368E-4
equation value using only low percentage data	2.978E-4

Visual inspection of the data and trend line show good agreement. This is also confirmed by the R^2 values (0.9414 and 0.9388) which are close to 1.0. The EAB dispersion factor will be 2.98E-4.

The LPZ dispersion factor output was also evaluated using a spreadsheet. The equation value, using all data, was higher than the value generated by KR PAVAN and the equation value using only the low probability data was lower than the value generated by KR PAVAN. Since a high degree of accuracy is not needed for the LPZ values and the KR PAVAN value is reasonable, the code values will be used for the LPZ.

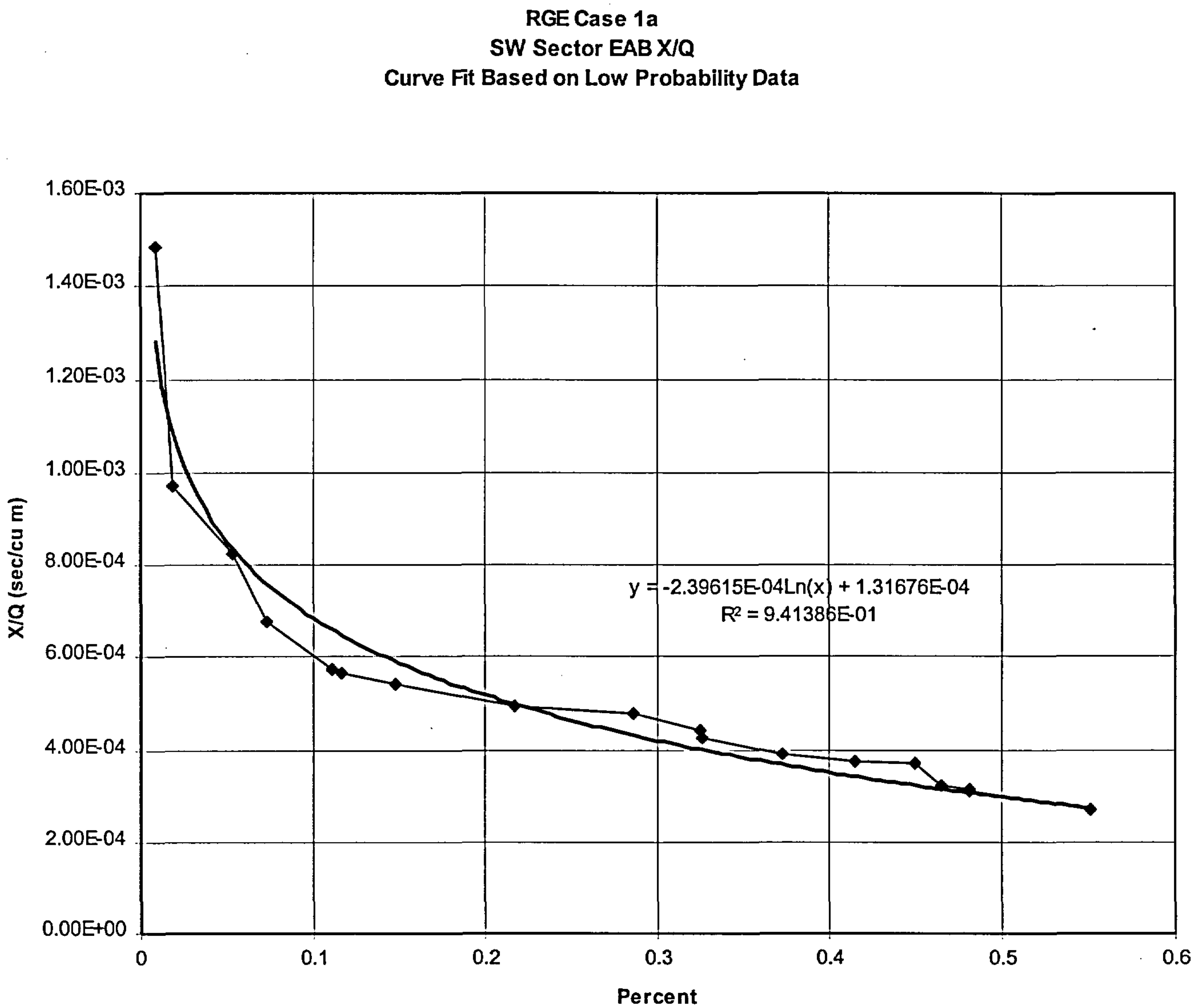
The X/Q values (sec/m^3) are:

Boundary	0-2 hr	0-8 hr	8-24 hr	24-96 hr	96-720 hr
EAB	2.98E-4	-	-	-	-
LPZ	-	2.29E-5	1.62E-5	7.59E-6	2.57E-6

TABLE 2.8	
KR PAVAN	INPUTS
Meteorological Data EAB distances, 16 wind speed directions (m)	Years 1992, 1993, 1994
S	450
SSW	450
SW	503
WSW	915
W	945
WNW	701
NW	1000
NNW	1000
N	1000
NNE	1000
NE	1000
ENE	1000
E	747
ESE	640
SE	503
SSE	450
LPZ distances (m)	4827
Wind speed considered to be calm (m/sec)	≤0.5
Activity releases	ground level
Height of wind speed measurement (m)	10
Calm hours	input separately from joint frequency distribution

Building - wake (m^2)	1071
Wind speed categories	14
Terrain adjustment factors	default

Figure 2.8.1 - Spreadsheet Analysis of Low Probability EAB X/Q Data



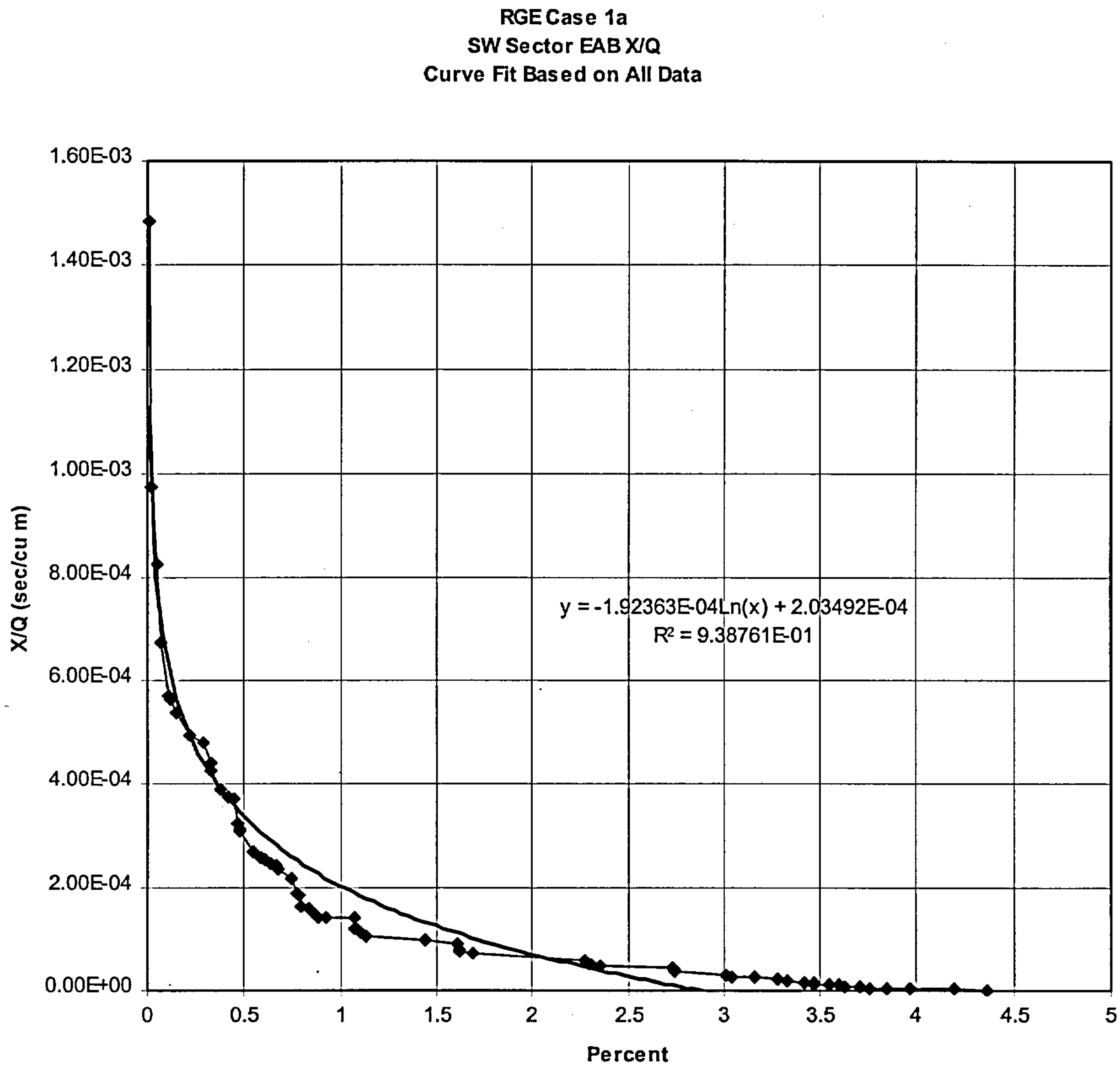
The 0.5 percent EAB value:

$$y := \left(-2.39615 \cdot 10^{-4}\right) \cdot \ln(x) + 1.31676 \cdot 10^{-4}$$

$$y = 2.97764 \times 10^{-4}$$

The resulting X/Q is rounded to 2.98E-4 sec/m^3.

Figure 2.8.2 - Spreadsheet Analysis of All EAB X/Q Data



The 0.5 percent EAB value:

$y{:=}-1.92363\cdot 10^{-4}\cdot \ln(x) + 2.03492\cdot 10^{-4}$

$y = 3.36828 \times 10^{-4}$

The resulting X/Q is rounded to 3.37E-4 sec/m^3

3.0 Iodine Spiking

For events where no fuel failure is postulated, iodine spiking is assumed. Two cases of iodine spiking are considered.

1. Accident Initiated Spike
2. Pre-Accident Spike

3.1 Accident Initiated Spike

The primary system transient causes an iodine spike in the primary system. The appearance rate is based on an equilibrium concentration of 1.0 μCi/gm Dose Equivalent I-131. The rate of increase and duration of the spike is event dependent. The following inputs are used in the calculation of the appearance rate.

TABLE 3.1 ACCIDENT INITIATED SPIKE INPUTS AND RESULTS	
Reactor coolant system volume, ft^3 rcs pzr (nominal minus 5% uncertainty)	 5506 436
Letdown purification flow rate, gpm	60 + 10%
Reactor coolant iodine concentrations @1 μCi/gram of DE 1-131, μCi/gram	I131 0.786 I132 4.54 E-3 I133 0.192 I134 1.55 E-4 I135 0.018
Mixed-bed demineralizer DF	100
Identified primary coolant leak rate, gpm	10
Unidentified primary coolant leak rate, gpm	1
Primary-to-secondary leak rate, gpd per SG	150
Letdown conditions Pressure, psia Temperature, °F	 15 127
Reactor coolant conditions Pressure, psia Temperature, °F	 2250 559
Appearance rates, Ci/hr I-131 I-132 I-133 I-134 I-135	 1.39E+1 2.49E-1 4.10E+0 1.79E-2 5.39E-1

3.2 Pre-Accident Spike - This assumes a transient has occurred prior to the event and has raised the primary coolant iodine concentration to the maximum full power value. This analysis assumes a value of 60 μCi/gm DE I-131. The resulting concentrations and inventories are:

I-131	4.71 E+1 μCi/gm	5.88 E+3Ci
I-132	2.72 E-1	3.39 E+1
I-133	1.15 E+1	1.43 E+3
I-134	9.32 E-3	1.16 E+0
I-135	1.07 E+0	1.33 E+2

4.0 General Discussion

4.1 The control room dose calculations use the same X/Q for both pre-isolated outside air and unfiltered inleakage. Pre-isolated outside air is all from the control room intake. Ginna does not have dual air intakes. Unfiltered inleakage may come from doors, penetrations into the control room envelope, air recirculation/filtration equipment, etc. The source to leakage location for all possible leak points is through other structures first (resulting in torturous paths) or longer source-to-receptor distances. Thus, the leakage point specific X/Q would be greater than that for the control room intake. The control room intake X/Q is assumed to be bounding for all control room dose calculations.

4.2 The nuclide data base used for all calculations is from ORIGEN2 (Reference 12). The nuclides are for a plant-specific representative 18 Month Fuel Cycle at end of life. The dose significant nuclide concentrations have been slightly increased to produce bounding doses.

4.3 All dose calculations assume the FGR11 and FGR12 dose conversion factors (References 10 and 11).

4.4 No credit is taken for elemental or methyl iodine removal inside containment by charcoal filters. This is indicated by assuming 0% efficiency as an input parameter. Credit is taken for particulate removal. Particulate removal is done by the inside containment HFPA filters.

4.5 Filter Loading - The RADTRAD code was used to calculate the inside containment HEPA filter particulate loading. The calculation was done for the conditions associated with a LBLOCA. The calculation assumed the filters operate for the duration of the calculation (720 hr.) which essentially removed all particulates from containment. The filter loading was approximately 1 oz/ft^2 which is judged to be well within the holding capability of the filters.

5.0 Loss-of-Coolant-Accident

5.1 Analysis

The analysis uses the alternate source term (AST) as defined in Reg. Guide 1.183 (Reference 5). The AST assumptions are listed on Table 5.1 and are consistent with Reg. Guide 1.183. The analysis is performed with the HABIT code version 1.1 (Reference 6) and the nuclide data base discussed in Section 4.2. The LBLOCA analysis consists of two parts: 1) Containment Leakage and 2) ECCS continuous leakage outside Containment. The resulting doses are summarized on Table 5.4

The airborne fraction (flashing fraction) used in the analysis is piece-wise time dependent and bounds the values based on sump (ECCS leakage) temperature from a Ginna-specific calculation. The values used in the analysis are illustrated on Figure 5.1.

The flashing fraction is estimated as follows:

$$FF = \frac{H_{exit} - H_l}{H_v - H_l}$$

Where:

FF = flashing fraction
H_{exit} = enthalpy of the relieved fluid (sump conditions)
H_l = enthalpy of liquid at 15 psia, saturated
H_v = enthalpy of vapor at 15 psia, 212°F.

Sump water temperature varies from 260°F at 1 hr. into the LOCA to 170°F at 50 hr. Sump pH is maintained greater than 7.0 on recirculation.

To determine the airborne fraction, a number of points were selected along the flashing curve, and then the curve was converted into a conservative step function. The value of each step is approximately 0.01 above the calculated flashed fraction. Even though the curve predicts that the flashed fraction goes to 0 at about 15 hours, the minimum airborne fraction is maintained at 0.01 out to 720 hours (only 25 hours shown in Fig 5.1). This is done to account for some droplet atomization.

Although these values are not as conservative as the fixed value of 10% suggested in the SRP, they are consistent with the intent of the SRP which is to use a conservative approximation.

5.2 Assumptions

A Large Break Loss of Coolant Accident (LBLOCA) occurs inside Containment.

One train of emergency power is assumed to fail. This results in only one train of Containment Recirculation Fan Coolers (CRFCs) operating and one train of Containment Spray.

At 52 minutes Containment Spray is stopped and sump recirculation is started and continues for the duration of the calculation.

At 4 hours the CRFCs are arbitrarily stopped, terminating particulate removal by filtration.

The Control Room parameters are listed on Table 5.3.

The Control Room is assumed isolated at 60 seconds and CREATS is up and operating at 70 seconds. Isolation from the radiation monitors and/or safety injection would occur well before the 60 seconds assumed in the analysis.

A passive ECCS failure of 50 gpm as identified in the Ginna UFSAR is not assumed in this analysis. However, the ECCS leakage has been increased to 4 gph.

The analysis uses the source term parameters in Table 5.1 and the Containment leakage parameters on Table 5.2.

Control room parameters are shown in Table 5.3 and 5.4.

ECCS Leakage - The analysis assumes a continuous leakage of 4 gph.

5.3 Results

The results are provided in Table 5.5.

FIGURE 5.1 - AIRBORNE FRACTION

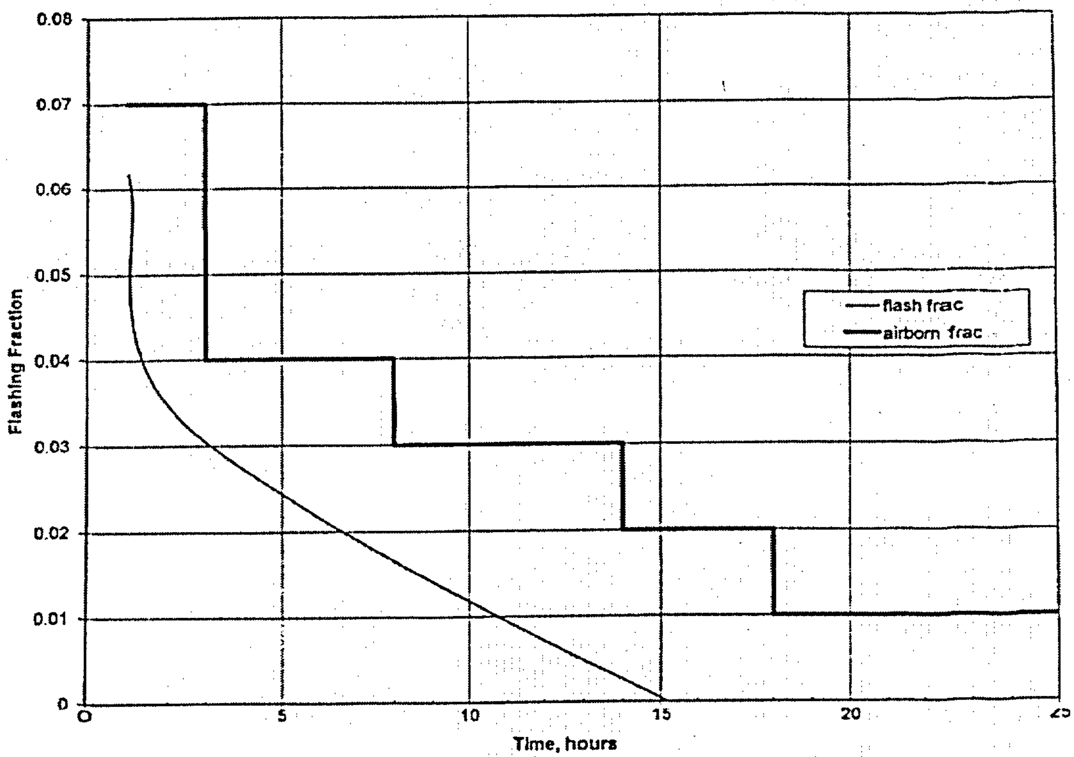


TABLE 5.1 ALTERNATE SOURCE TERM (REFERENCE 5) Core Inventory Fraction Released Into Containment			
Nuclide Group	**Gap Release Phase**	**Early In-Vessel Phase**	**Total[4]**
Halogens	0.05	0.35	0.4
Noble Gases	0.05	0.95	1.0
Alkali Metals	0.05	0.25	0.3
Tellurium	0	0.05	0.05
Ba, Sr	0	0.02	0.02
Noble Metals	0	0.0025	0.0025
Cerium	0	0.0005	0.0005
Lanthanides	0	0.0002	0.0002

TABLE 5.1 ALTERNATE SOURCE TERM (REFERENCE 5) Timing of LOCA Core Inventory Release Phases		
Release Phase	**Onset**	**Duration**
Gap Release	30 sec	0.5 hr[5]
Early In-Vessel	0.5 hr	1.3 hr

TABLE 5.1 ALTERNATE SOURCE TERM (REFERENCE 5) Nuclide Groups	
Halogens	I
Noble Gases	Kr, Xe

[4]Fractions apply to both containment and ECCS leakage

[5]The duration of the gap release, specified in Reference 5, is 0.5 hr. The specified start of the gap release is 30 seconds and the end of the release is 0.5 hr. Thus, the duration of the gap release is modeled as 0.5 hr - 30 sec = 0.492 hr, rather than 0.5 hr.

TABLE 5.1 ALTERNATE SOURCE TERM (REFERENCE 5) Nuclide Groups	
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

TABLE 5.1 ALTERNATE SOURCE TERM (REFERENCE 5) Nuclide Composition, fraction		
Form	In Containment Atmosphere	In ECC Solution
Iodine elemental methyl particulate	 0.0485 0.0015 0.95	 0.97 0.03 0
All other nuclides particulate	 1.0	 1.0

TABLE 5.2 CONTAINMENT/ECCS LEAKAGE PARAMETERS	
Parameter	**Value**
Reactor power, Mwt(including 2% uncertainty)	1550
Containment net free volume, ft3	1.0E6
Containment sprayed fraction	0.78
Containment leak rate, %/day 0-24 hours > 24 hours	 0.2 0.1
Containment fan cooler flow and operation number of operating units (per train) flow rate per unit, cfm total filtered flow rate, cfm HEPA (2 units) initiation delay, sec. termination of iodine removal, hours	 2 30,000 60,000 50 4
Containment fan cooler iodine removal efficiency, % Elemental Methyl Particulate	 90 50 95
Containment injection spray flow rate, gpm (per train) initiation delay, sec termination (end of spray injection), min	 1300 80 52
Iodine and particulate removal by spray, hr-1 elemental particulate	 20 3.5[1]
Containment sump volume, ft^3	264,700

[1]Represents the 10th percentile value from the Powers model (Reference 7).

TABLE 5.2 CONTAINMENT/ECCS LEAKAGE PARAMETERS	
Parameter	**Value**
ECCS leakage	
Continuous leakage rate, gal/hr	4
Start time, hr	1
Termination time, hr	720
Airborne fraction	
0-3 hr	0.07
3-8 hr	0.04
8-14 hr	0.03
14-18 hr	0.02
>18 hr	0.01
Atmospheric dispersion X/Q, sec/m^3	
EAB 0-2 hr	4.8E-4
LPZ 0-8 hr	3.0E-5
8-24 hr	2.1E-5
24-96 hr	8.6E-6
96-720 hr	2.5E-6
Breathing rates, m^3/sec	
EAB & LPZ 0-8 hr	3.47E-4
8-24 hr	1.75E-4
24-720 hr	2.32E-4

TABLE 5.3 CONTROL ROOM PARAMETERS		
Parameter	**Value**	
Habitable volume, ft^3	36,211	
Normal Operating Mode make-up air flow rate, cfm	2000+10%	
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm Unfiltered in-leakage, cfm	 90 70 98 6000-10% 300	
Breathing rate, m^3/sec	3.47E-4	
Occupancy factors 0-24 hr 24-96 hr 96-720 hr	 1 0.6 0.4	
Atmospheric dispersion X/Q sec/m^3	Containment Leakage	ECCS Leakage
0-2 hr	1.57E-3	3.89E-3
2-8 hr	1.12E-3	2.99E-3
8-24 hr	4.47E-4	9.63E-4
24-96 hr	3.69E-4	8.98E-4
96-720 hr	3.10E-4	8.23E-4

Table 5.4 Flow Rate and Iodine Removal Schedule				
	Inleakage		Recirculation	
Time, hours	cfm	iodine removal efficiency, %[1]	cfm	iodine removal efficiency, %[1]
0-0.0167[2]	2200	0/0/0	0	0/0/0
[3]0.0167-0.0194	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

TABLE 5.5 LBLOCA DOSE SUMMARY, REM TEDE			
	EAB Max. 2-hour	LPZ 720 hour	Control Room 720 hour
Containment Leakage	5.447	0.918	2.062
ECCS Leakage	0.475	0.145	0.968
Total	5.92	1.06	3.03
Acceptance Criteria	25	25	5

[1]Elemental/Methyl/Particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

6.0 Fuel Handling Accident

6.1 Analysis

This calculation determines the offsite and Control Room doses (TEDE) for a fuel handling accident (FHA). The analysis uses the alternate source term and accompanying TEDE methodology and conservative control room X/Q values that are calculated with the ARCON96 code. Two cases will be evaluated:

- FHA inside Containment
- FHA in the Spent Fuel Pool

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base as discussed in Section 4.2 are used. Since the release from the FHA is assumed to end after 2 hours, the dose calculations are terminated after all contributions are accounted for; 2 hours for LPZ and 24 hours for the Control Room. The resulting doses are presented on Table 6.4.

6.2 Assumptions

Both cases assume the fuel rods in one fuel assembly fail.

Activity from the damaged fuel rods is assumed to be instantaneously released to the pool water.

There is a minimum of 23 feet of water above the fuel.

The activity release rate is independent of the actual ventilation flow rate. The activity release rate is adjusted to ensure all radioactive material that escapes from the reactor cavity or spent fuel pool is released to the environment over a two hour period.

The activity from a FHA in Containment is assumed to be released from Containment to the environment via the perimeter seals of the Equipment Hatch roll-up door. No filtration or absorption of iodine is assumed.

The activity from a FHA in the spent fuel pool is assumed to be released from the pool area to the environment via the plant vent. The dose conversion factors from FGR11 and 12 are used (References 10 and 11).

Note that the charcoal filter system for the spent fuel pool area is not ESF or safety-related and the charcoal filter system would be unavailable if a coincident loss of offsite power were to occur. The Technical Specifications require use of the system during irradiated fuel movement within the Auxiliary Building to minimize doses. Therefore, the system is credited in the dose analysis.

The FHA dose analysis assumptions are listed on Table 6.1. The Control Room assumptions are listed on Table 6.2.

The Control Room is assumed to be isolated within 60 seconds via the radiation monitors. A comparison of the nuclide concentration in the Control Room intake for the FHA to the radiation monitor response showed a Control Room isolation signal would occur before the 60 seconds assumed in the calculation.

Fission product inventories and activities released from the SFP are shown in Table 6.3.

TABLE 6.1 FHA DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Reactor power, Mwt(including 2% uncertainty)	1550
Power Peaking Factor	1.75
Number of damaged fuel assemblies	1
Fission product inventory in damaged assemblies after decay	Values shown in Table 6.3
Time after reactor shutdown, hr	100
Fuel rod gap fractions I-131 other halogens Kr-85 other noble gases alkali metals	 0.08 0.05 0.1 0.05 0.12
Iodine species above water elemental iodine organic iodide	 0.57 0.43
Pool DF elemental iodine organic iodide particulate Overall Pool DF	 500 1 ∞ 200
Containment net free volume, ft^3	1E6
Exhaust flow rate, cfm	7.68E4
Duration of activity release, hr	2
Iodine removal efficiency Containment (all iodine forms) Fuel Pool elemental iodine organic iodide	 0 0.9 0.7

TABLE 6.1 FHA DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Atmospheric dispersion, X/Q, sec/m^3 EAB 0-2 hr LPZ 0-8 hr	 4.8 E-4 3.0 E-5
Breathing rate, m^3/sec EAB & LPZ 0-8 hr	 3.47 E-4

TABLE 6.2 CONTROL ROOM PARAMETERS		
Habitable volume, ft^3		36,211
Normal Operating Mode make-up air flow rate, cfm		 2000+10%
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate Flow rate, cfm Unfiltered in-leakage, cfm		 90 70 98 6000-10% 300
Breathing rate, m^3/sec		3.47 E-4
Occupancy factor 0-24 hr 24-96 hr 96-720 hr		 1 0.6 0.4
Atmospheric dispersion, X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr	FHA Containment 5.64 E-3 4.69 E-3 1.66 E-3	FHA Spent Fuel Pool 1.79 E-3 1.15 E-3 4.95 E-4

Flow Rate and Iodine Removal Schedule				
	Inleakage		Recirculation	
Time, hours	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0 - 0.0167[2]	2200	0/0/0	0	0/0/0
[3]0.0167 - 0.0194	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

[1]Elemental/Methyl/Particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

TABLE 6.3 FISSION PRODUCT INVENTORY AND ACTIVITY RELEASED FROM POOL						
Nuclide	Total Core Activity - 100 hours decay, Ci(A)	Core Damage Fraction (F)	Gap Fraction (G)	Peaking Factor (P)	Overall Pool DF	Activity Released from Pool, Ci (A)
I-131	2.98E+07	0.008264	0.08	1.75	200	1.76E+02
I-132	2.52E+07	0.008264	0.05	1.75	200	9.29E+01
I-133	3.12E+06	0.008264	0.05	1.75	200	1.15E+01
I-134	0.00E+00	0.008264	0.05	1.75	200	0.00E+00
I-135	2.23E+03	0.008264	0.05	1.75	200	8.22E-03
Kr-85m	2.15E+00	0.008264	0.05	1.75	1	1.55E-03
Kr-85	4.98E+05	0.008264	0.1	1.75	1	7.20E+02
Kr-87	4.58E-17	0.008264	0.05	1.75	1	3.31E-20
Kr-88	7.48E-04	0.008264	0.05	1.75	1	5.41E-07
Xe-131m	4.42E+05	0.008264	0.05	1.75	1	3.20E+02
Xe-133m	1.10E+060	0.008264	0.05	1.75	1	7.95E+02
Xe-133	5.71E+07	0.008264	0.05	1.75	1	4.13E+04
Xe-135m	3.57E+02	0.008264	0.05	1.75	1	2.58E-01
Xe-135	1.09E+05	0.008264	0.05	1.75	1	7.88E+01

Core damage fraction is 1/121 = 0.008264. The total number of fuel assemblies in the core is 121.

The activity released from the pool (A) is calculated as follows: Note 2% added to iodine in Section 4.2.

$$A = \frac{A_c * F * G * P}{DF}$$

TABLE 6.4 FHA, DOSE, REM, TEDE			
	EAB Max - 2 hr	LPZ, 2 hr	Control Room, 24 hr
FHA - inside Containment via roll-up door	1.12	0.07	1.18
FHA - Spent Fuel Pool	0.31	0.019	0.089
Acceptance Criteria	6.3	6.3	5

7.0 Main Steam Line Break

7.1 Analysis

This calculation determines the offsite and Control Room doses (TEDE) for the Main Steam Line Break (MSLB) outside the Containment. The analysis uses the alternate source term and the accompanying TEDE methodology and conservative control room X/Q values that are calculated with the ARCON96 code. The MSLB analysis includes the following cases:

- MSLB with accident initiated iodine spike
- MSLB with pre-accident iodine spike

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base as discussed in Section 4.2 are used. No fuel failures are postulated for the MSLB.

7.2 Assumptions

The purpose of this analysis is to calculate the steam releases from the faulted and intact Steam Generator (SG) during a steam line break to the atmosphere. Therefore, breaks inside Containment are non-applicable.

Because of an augmented inspection program, breaks between the Containment penetrations and inside the Intermediate Building are limited to connection pipes only with the largest pipe being 6" (UFSAR Section 3.6.2.4.5.2). Larger pipe breaks can only be postulated downstream of the Intermediate Building, i.e., inside the Turbine Building. Therefore, the break is assumed to occur in the 36" header inside the Turbine Building. This is the largest pipe break that can occur outside Containment. The break area is limited to 1.4 ft^2 because of a flow restrictor in the SG outlet nozzle.

The scenario consists of a header break. The single failure is assumed to be a failure of the main steam isolation valve on the faulted SG. Initially the break is fed by both SGs. Following steam line isolation, the break is fed only by the faulted SG. At approximately 10 minutes the faulted SG is isolated by operator action. The intact SG is then used for cooldown, where steam is released to the atmosphere through the intact SG Atmospheric Relief Valve (assumed to be 8 hr.) until the releases are stopped.

A primary to secondary leakage of one gpm to each SG is assumed for the duration of the event (8 hr). The faulted SG is assumed to be dry at 10 minutes. and remain dry for the event. The intact SG is isolated from the break within the first minute and auxiliary feedwater maintains SG level for the duration of the event.

All of the initial iodine inventory in the faulted SG is assumed released to the environment by 10 minutes. The iodine from the primary-to-secondary leakage into the faulted SG is released directly to the environment with no credit for retention. The initial iodine inventory in the intact SG is mixed with the primary-to-secondary leakage into the SG and released to the environment assuming an iodine partition of 100. The steam release from the intact SG is based on a LOFTRAN simulation of the MSLB followed by an energy balance to simulate the cooldown to RHR conditions. All noble gas activity carried over to the SGs is assumed to be immediately released to the environment.

Initially the Control Room HVAC is operating normally with a nominal 2000 cfm of makeup air. Isolation is assumed to occur at 60 sec and CREATS is operating at 70 sec assuming a nominal 6000 cfm recirculation flow. Since isolation is caused by a safety injection signal, the Control Room would be isolated well before the 60 sec. assumed in the analysis. Following isolation, 300 cfm of unfiltered inleakage is assumed for the duration of the calculation.

The releases from the steam break are assumed to stop at 8 hr. The Control Room calculation is continued until 24 hr to ensure all dose contributions are accounted for.

Accident - Initiated Iodine Spike: A spike factor of 500 with a duration of 8 hours is assumed. The initial appearance rates are listed on Table 3.1.

Pre-Accident Iodine Spike: The iodine concentrations are based on 60 μCi/gm DE I-131 and listed in Section 3.2.

Additional assumptions are listed in Table 7.1.

The Control Room parameters are listed on Table 7.2 and 7.3.

7.3 Results

The results for the MSLB are shown in Table 7.4.

TABLE 7.1 MSLB DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Reactor power, Mwt(including 2% uncertainty)	1550
Initial reactor coolant activity, pre-accident iodine spike iodine μCi/gm of D.E. I-131 noble gas fuel defect level, %	 60 1.0
Initial reactor coolant activity, accident initiated iodine spike iodine μCi/gm of D.E. I-131 noble gas fuel defect level, %	 1.0 1.0
Concurrent iodine spike factor	500
Duration of concurrent iodine spike, hours	8
Initial secondary coolant iodine activity iodine μCi/gm of D.E. I-131 Concentration Ci	 0.1 I-131 4.57 E+0 I-132 2.64 E-2 I-133 1.12 E+0 I-134 9.04 E-4 I-135 1.03 E-1
Primary-to-secondary leakage (post accident) to SGs leak rate (cold conditions) per SG, gpm duration of leakage, hours	 1 8
Mass of primary coolant, gm	1.247 E+8
Initial mass of secondary coolant, gm faulted SG intact SG	 5.817 E+7 5.817 E+7

TABLE 7.1 MSLB DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Steam Releases faulted SG 0 - 610 sec 610 sec - 8 hr intact SG 0 - 610 sec 610 sec - 8 hr	 128,237 lb 0 lb 37,780 lb 755,097 lb
Primary to Secondary Leakage	1 gpm per SG
Steam generator iodine partition coefficients (mass-based) Activity release from faulted SG elemental methyl Activity release from intact SG elemental methyl Noble gas, all SG	 1 1 100 1 1
Iodine fractions assumed in the reactor coolant and SG water elemental iodine organic iodide	 0.97 0.03
Atmospheric dispersion X/Q sec/m^3 EAB 0-2 hr LPZ 0-8 hr	 4.8E-4 3.0E-5
Breathing rate m^3/sec EAB & LPZ 0-8 hr 8-24 hr 24-720 hr	 3.47 E-4 1.75 E-4 2.32 E-4

TABLE 7.2 CONTROL ROOM PARAMETERS	
Parameter	**Value**
Habitable volume, ft^3	36,211
Normal Operating Mode make-up air flow rate, cfm	 2000+10%
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm Unfiltered in-leakage, cfm	 90 70 98 6000-10% 300
Breathing rate, m^3/sec	3.47 E-4
Occupancy factor 0-24 hr 24-96 hr 96-720 hr	 1 0.6 0.4
Atmospheric dispersion, X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr	 2.57 E-3 1.92 E-3 8.08 E-4

Table 7.3 Flow Rate and Iodine Removal Schedule				
Time, hours	Inleakage		Recirculation	
	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0 - 0.0167[2]	2200	0/0/0	0	0/0/0
[3]0.0167 - 0.0194	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

TABLE 7.4 RESULTS FOR MAIN STEAM LINE BREAK			
	EAB Max - 2 hr TEDE	LPZ, 8 hr TEDE	Control Room 24 hr TEDE
Accident Initiated Iodine Spike	1.05	0.15	0.64
Acceptance Criteria	2.5	2.5	5
Pre-Accident Iodine Spike	0.15	0.03	0.18
Acceptance Criteria	25	25	5

[1]Elemental/Methyl/particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

8.0 Steam Generator Tube Rupture (SGTR)

8.1 Analysis

This calculation determines the offsite and Control Room doses for the SGTR accident. The analysis uses alternate source term and accompanying TEDE methodology and conservative Control Room X/Q values, that are calculated with the ARCON96 code. The SGTR analysis includes the following cases:

- SGTR with accident-initiated spike
- SGTR with pre-accident iodine spike

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base discussed in Section 4.2 are used.

8.2 Assumptions

Input parameters are listed in Table 8.1 below.

The break flow and steam release data for the ruptured SG, and steam release data for the intact SG is taken from the analysis discussed in Section 15.6 of Reference 3 and listed on Table 8.2.

The Control Room parameters are listed on Table 8.3.

Control Room isolation is assumed at 6 minutes which bounds the safety injection signal generation time for the Reference 3, Section 15.6 SGTR. The ARV is the source point for the Control Room X/Q.

Accident-Initiated Iodine Spike:

The initial appearance rates are listed on Table 3.1. The input parameters are listed on Table 8.1 and the results are presented on Table 8.5. The dose calculations are terminated after all dose contributions are accounted for.

Pre-Accident Iodine Spike:

The iodine concentrations are based on 60 μCi/gm DE I-131 and listed in Section 3.2. The input parameters are listed on Table 8.1 and results are presented on Table 8.5. The dose calculations are terminated after all dose contributions are accounted for.

TABLE 8.1 SGTR DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Reactor power, MwT(including 2% uncertainty)	1550
Initial reactor coolant activity, pre-accident iodine spike iodine, µCi/gm of DE I-131 noble gas fuel defect level, %	 60 1.0
Initial reactor coolant activity, accident initiated iodine spike iodine, ci/gm of DE I-131 noble gas fuel defect level, %	 1.0 1.0
Concurrent iodine spike factor	335
Duration of concurrent iodine spike, hours	8
Initial secondary coolant iodine activity, µCi/gm of DE I-131	0.1
Primary-to-secondary leakage to intact SG leak rate (cold conditions) duration of leakage, hours	 150 gal/day 8
Mass of primary coolant, gm	1.247×10^8
Initial mass of secondary coolant, gm faulted SG intact SG	 3.27×10^7 3.27×10^7
Steam generator elemental iodine partition coefficients (mass-based) Activity release from faulted SG via boiling of bulk water via flashed break flow Activity release from intact SG	 100 1.0 100
Steam generator partition coefficient for organic iodide and noble gas release	1.0
Iodine species assumed in the reactor coolant and SG water elemental iodine organic iodide	 0.97 0.03

TABLE 8.1 SGTR DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Atmospheric dispersion, X/Q, sec/m^3 EAB 0-2 hr LPZ 0-8 hr	 4.8 E-4 3.0 E-5
Breathing Rates, m^3/sec EAB & LPZ 0-8 hr 8-24 hr	 3.47E-4 1.75E-5

Table 8.2 Steam Releases and Rupture Flow				
	Time periods, seconds			
Mass, 1000 lb_m	0-Trip	Trip-break	Break - 2 hours	2 hrs - RHR
Ruptured SG to: Condenser[1] Atmosphere	 45.5 -	 - 62.4	 - 0	 - 31.6
Intact SG to: Condenser Atmosphere	 45.2 -	 - 60.0	 - 147.5	 - 459.9
Rupture flow	2.9	107.4	-	-

trip: Reactor trip (49 seconds).
break: SG and RC pressures are equal, rupture flow is terminated (3492. sec.).
RHR: RHR operating conditions are achieved, steaming to the environment is terminated (8 hours).

[1]The analysis conservatively treats steam released to the condenser the same as a direct release to the atmosphere, i.e., elemental iodine partition is 100.

TABLE 8.3 CONTROL ROOM PARAMETERS	
Parameter	**Value**
Habitable volume, ft^3	36,211
Normal Operating Mode make-up air flow rate, cfm	 2000+10%
Accident Operating Mode recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm unfiltered in-leakage, cfm	 90 70 98 6000-10% 300
Breathing rate, m^3/sec	3.47E-4
Occupancy factor 0-24 hr 24-96 hr 96-720 hr	 1 0.6 0.4
Atmospheric dispersion, X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr	 3.66E-3 2.49E-3 1.07E-3

Table 8.4 **Flow Rate and Iodine Removal Schedule**				
Time, hours	Inleakage		Recirculation	
	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0-0.1[2]	2200	0/0/0	0	0/0/0
[3]0.1-0.103	300	0/0/0	0	0/0/0
>0.103	300	0/0/0	5400	90/70/98

TABLE 8.5 **RESULTS FOR SGTR**			
	EAB Max 2 hr	LPZ, 8 hr	Control Room 24 hr
Accident Initiated Iodine Spike (TEDE)	0.22	0.017	0.14
Acceptance Criteria	2.5	2.5	5
Pre-Accident Iodine Spike (TEDE)	0.71	0.051	0.88
Acceptance Criteria	25	25	5

[1]Elemental/Methyl/Particulate

[2]0 to 360 seconds

[3]360 to 370 seconds

9.0 Locked Rotor Accident

This calculation determines the offsite and Control Room doses for the LR accident. The analysis uses alternate source term and accompanying TEDE methodology and conservative Control Room X/Q values, that are calculated with the ARCON96 code. The LR analysis includes the following case:

- Primary-to-secondary leakage with SG activity releases

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base discussed in Section 4.2 are used.

9.1 Assumptions

Input parameters are listed in Table 9.1 and 9.2 below.

It is conservatively assumed 100% of the fuel rods experience DNB and are therefore assumed to release their gap activity into the reactor coolant system.

The initial reactor coolant iodine activity is based on a pre-accident spike discussed in Section 3.2. The concentrations are based on 60 uCi/gm of DE I-131. The noble gas activity is based on 1% fuel defects.

The initial secondary coolant iodine activity is based on 0.1 uCi of DE I-131.

The assumed post-accident primary-to-secondary leak rate is 500 gal/day per SG. This bounds the current limit of 144 gpd/SG and a future Technical Specification limit of 150 gpd/SG.

A partition coefficient of 100 is assumed for elemental iodine in the secondary coolant. No partitioning is assumed for organic iodine or noble gas. No particulates are assumed to be released to the atmosphere with the secondary side steam.

The steam release from the SGs is based on a LOFTRAN simulation of the LR followed by an energy balance to simulate the cooldown to RHR conditions. RHR System is assumed to be placed into service for heat removal 8 hours after the initiation of the LR.

Initially the Control Room HVAC is operating normally with a nominal 2000 cfm of makeup air. Isolation is assumed to occur at 60 sec. via the radiation monitors. A comparison of the nuclide concentration in the Control Room intake for the LR to the radiation monitor response showed a Control Room isolation signal would occur before the 60 sec. assumed in the calculations. CREATS is assumed operating at 70 sec. assuming a nominal 6000 cfm recirculation flow.

The EAB and LPZ X/Qs are the new values calculated by K R PAVAN as discussed in Section 2.8.2.

TABLE 9.1 LR Dose Analysis Assumptions	
Parameter	**Value**
Reactor power, Mwt(including 2% uncertainty)	1550
Failed Fuel, %	100
Initial reactor coolant activity, pre-accident iodine spike iodine, uCi/gm of DE I-131 noble gas fuel defect level, %	 60 1.0
Initial secondary coolant iodine activity, uCi/gm of DE I-131	0.1
Primary-to-secondary leakage (post accident) to SGs leak rate (cold conditions per SG, gpd duration of leakage, hours	 500 8
Mass of primary coolant, gm	$1.247\text{x}10^{8}$
Initial mass of secondary coolant in 2 SGs, gm	8.501E+7
Steam Releases (2 SGs), lb 0-10 min. 10-30 min. 0.5-8 hr.	 54,620 14,446 685,229
Steam generator iodine partition coefficients (mass-based) elemental methyl (organic)	 100 1
Iodine fractions in the reactor coolant and SG water elemental iodine methyl (organic) iodide	 0.97 0.03
Atmospheric dispersion X/Q sec/m^3 EAB 0-2 hr LPZ 0-8 hr	 2.98E-4 2.29E-5

TABLE 9.1 LR Dose Analysis Assumptions	
Parameter	**Value**
Breathing rate m^3/sec EAB & LPZ 0-8 hr 8-24 hr	 3.47E-4 1.75E-4

TABLE 9.2 CONTROL ROOM PARAMETERS	
Parameter	**Value**
Habitable volume, ft^3	36,211
Normal Operating Mode make-up air flow rate, cfm	 2000+10%
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm Unfiltered in-leakage, cfm	 90 70 98 6000-10% 300
Breathing rate, m^3/sec	3.47 E-4
Occupancy factor 0-24 hr 24-96 hr 96-720 hr	 1 0.6 0.4
Atmospheric dispersion, X/Q, sec/m^3 0-2 hr 2-8 hr 8-24 hr	 3.66 E-3 2.49 E-3 1.07 E-3

Table 9.3 Flow Rate and Iodine Removal Schedule				
	Inleakage		Recirculation	
Time, hours	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0 - 0.0167[2]	2200	0/0/0	0	0/0/0
[3]0.0167 - 0.0194	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

TABLE 9.4 RESULTS FOR LOCKED ROTOR			
	EAB Max - 2 hr TEDE	LPZ, 8 hr TEDE	Control Room 24 hr TEDE
Elemental iodide	1.150	0.209	1.370
Methyl iodide	1.022	0.254	2.115
Noble gas	0.578	0.090	0.232
Total	2.750	0.553	3.717
Acceptance criteria	2.5	2.5	5

[1]Elemental/Methyl/particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

10.0 Rod Ejection Accident

This calculation determines the offsite and Control Room doses (TEDE) for Rod Ejection Accident (REA). The analysis uses the alternate source term and the accompanying TEDE methodology and conservative control room X/Q values that are calculated with the ARCON96 code. The REA analysis includes the following cases:

- Containment leakage
- Primary-to-secondary leakage with SG activity release.

Doses are calculated for the following receptors:

- Exclusion Area Boundary (EAB), maximum 2 hour dose
- Outer boundary of the Low Population Zone (LPZ), 30 day dose (8 hr for secondary side transport)
- Control Room, 30 day dose (24 hr for secondary side transport)

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base described in Section 4.2 are used. Ten percent of the core is assumed to fail. This is based on a Ginna specific calculation (Reference 3, Section 15.4.5.3.5). The release fraction used in the analysis is the product of the core damage, the peaking factor, and the gap fraction. The input parameters are listed on Table 10.1.

10.1 Containment Leakage

The containment leakage calculation assumes the gas activity is instantaneously released from the core to containment atmosphere. No credit is taken for removal of elemental or methyl iodine by the CRFC charcoal filters. The CRFCs only remove particulate iodine by the associated HEPA filters. The CRFCs are assumed to be operating at 53 seconds based on a 3 inch SBLOCA. The CRFCs are arbitrarily terminated after four hours since there is no longer a significant particulate concentration.

Containment spray is assumed not to actuate for the REA. Particulate removal is assumed by natural deposition. The removal coefficient is based on the correlations provided in Reference 8, Table 2.2.2.1-1. The first is for the time period 0 to 0.5 hr and the second is for 0.5 to 1.8 hr. The 10th percentile is the most conservative (smallest removal rate) and is used in this calculation. Only the smallest value is used and is held constant for the duration of the calculation.

10.2 Primary-to-Secondary Leakage

The initial reactor coolant iodine activity is based on a pre-accident spike discussed in Section 3.2. The concentrations are based on 60μCi/gm of DE I-131. The noble gas activity is based on 1% fuel defects. Gap (10% failed fuel rods) activity is released instantaneously and homogeneously mixed in the reactor coolant. The activity release fraction is the product of core damage, the peaking factor, and gap fraction.

The initial secondary coolant iodine activity is based on 0.1µi of DE I-131.

The assumed post-accident primary-to-secondary leak rate is 500 gal/day per SG. This bounds the current limit of 144 gpd/SG and a future Technical Specification limit of 150 gpd/SG.

A partition coefficient of 100 is assumed for elemental iodine in the secondary coolant. No partitioning is assumed for organic iodine or noble gas. No particulates are assumed to be released to the atmosphere with the secondary side steam.

The steam release from the SGs is based on a LOFTRAN simulation of the REA followed by an energy balance to simulate the cooldown to RHR conditions. RHR system is assumed to be placed into service for heat removal 8 hours after the initiation of the REA.

Initially the Control Room HVAC is operating normally with a nominal 2000 cfm of makeup air. Isolation is assumed to occur at 60 sec. via the radiation monitors. A comparison of the nuclide concentration in the Control Room intake for the REA to the radiation monitor response showed a Control Room isolation signal would occur before the 60 sec. assumed in the calculations. CREATS is assumed operating at 70 sec. assuming a nominal 6000 cfm recirculation flow.

TABLE 10.1 REA CONTAINMENT PARAMETERS	
Parameter	**Value**
Reactor power, MwT(including 2% uncertainty)	1550
Failed Fuel, % of core Gap fraction Peaking factor, fraction	10 0.10 1.75
Initial primary coolant activity iodine noble gas	 60µCi/gm of DE I-131 1% fuel defects
Iodine forms particulate elemental organic	 0.95 0.0485 0.0015
Containment net free volume, ft^3	10E6
Containment leak rate, %/day 0-24 hr >24 hr	 0.2 0.1
Containment fan cooler flow and operation number of operating units flow rate per unit, cfm total filtered flow rate, cfm HEPA (2 units) initiation delay CRFCs (HEPA) termination of particulate iodine removal, hours	 2 30,000 60,000 53 sec 4
Containment fan cooler iodine removal efficiency, % elemental methyl particulate	 0 0 95
Natural deposition coefficient, 1/hr	0.023

TABLE 10.1 REA CONTAINMENT PARAMETERS	
Parameter	**Value**
Atmospheric dispersion, X/Q, sec/m^3	
EAB 0-2 hr	4.8 E-4
LPZ 0-8 hr	3.0 E-5
8-24 hr	2.1 E-5
24-96 hr	8.6 E-6
96-720 hr	2.5 E-6
Breathing rate, m^3/sec	
EAB & LPZ	
0-8 hr	3.47 E-4
8-24 hr	1.75 E-4
24-720 hr	2.32 E-4

TABLE 10.2 PARAMETERS FOR REA SECONDARY SIDE ACTIVITY RELEASE	
Parameter	**Value**
.Reactor power, MwT(including 2% uncertainty)	1550
Failed fuel, % of core	10
gap fraction	0.10
peaking factor, fraction	1.75
Initial secondary coolant iodine activity, ci/gm of DE I-131	0.1
Primary-to-secondary leakage	
leak rate, gpd per SG	500
duration, hr	8
Mass of primary coolant, gm	1.247E8
Initial mass of secondary coolant, gm per 2 SGs	8.5E7
Steam released from S.S. to environment, gm/min	
0-10 min	2.478E6
10-30 min	3.276E5
30 min - 8 hr	6.907E5

TABLE 10.2 PARAMETERS FOR REA SECONDARY SIDE ACTIVITY RELEASE	
Steam generator iodine partition coefficient (mass-based) elemental methyl	 100 1
Iodine species assumed in the SG water elemental iodine methyl iodide	 0.97 0.03

TABLE 10.3 CONTROL ROOM PARAMETERS		
Habitable volume, ft^3		36,211
Normal operating Mode make-0up air flow rate, cfm		 2000+10%
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm Unfiltered in-leakage,cfm		 90 70 98 6000-10% 300
Breathing rate, m^3/sec		3.47E-4
Occupancy factors 0-24 hr 24-96 hr 96-720 hr		 1 0.6 0.4
Atmospheric dispersion X/Q, sec/m^3	Containment Leakage	ARV
0-2 hr	1.57E-3	3.66E-3
2-8 hr	1.12E-3	2.49E-3
8-24 hr	4.47E-4	1.07E-3
24-96 hr	3.69E-4	
96-720 hr	3.10E-4	

Table 10.4 Control Room Flow Rate and Iodine Removal Schedule for REA				
Time, hours	Inleakage		Recirculation	
	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0-0.0167[2]	2200	0/0/0	0	0/0/0
0.0167-0.0194[3]	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

TABLE 10.5 REA DOSE SUMMATION, rem, TEDE			
	EAB, max -2 hour	LPZ, 720 hours (CNMT), 8 hours (secondary side)	Control Room, 720 hours (CNMT), 24 hours (secondary side)
Containment Leakage	2.859E-01	4.825E-02	1.311E-01
Secondary Side, Elemental Iodine	4.539E-01	6.759E-02	3.196E-01
Secondary Side, Noble Gas	3.263E-01	4.125E-02	8.132E-02
Secondary Side, Methyl Iodide	4.032E-01	8.244E-02	5.043E-01
TOTAL	1.47E+00	2.40E-01	1.04E+00

[1]Elemental/Methyl/Particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

11.0 Tornado Missile in Spent Fuel Pool

11.1 This calculation determines the offsite and Control Room doses (TEDE) for a tornado missile accident (TMA). The analysis uses the alternate source term and accompanying TEDE methodology and conservative Control Room X/Q values calculated as discussed in Section 2.7.

The AST defined in Reference 5 is used. The HABIT code (Reference 6) and HABIT nuclide data base as discussed in Section 4.2 are used. The analysis assumes 9 fuel assemblies are damaged (5 fuel assemblies decayed for 100 hours and four fuel assemblies decayed for 60 days) based on the size of a telephone pole missile. The nuclide inventory in the damaged assemblies is estimated by applying a power peaking factor of 1.75 to the average assembly inventory. Activity from the damaged assemblies is assumed to be instantaneously released to the pool water. After applying decontamination factors of the pool water, the resulting elemental and organic fractions above the water are 0.57 and 0.43. The activity above the pool is assumed to be released to the environment. No iodine removal is assumed. Reference 5 suggests using a two-hour activity release. Since the duration of the tornado is uncertain, and may be less than two hours, two cases were run.

Case 1) All activity was released over two hours. The activity released over the first hour was at tornado conditions. The activity released over the second hour was at normal atmospheric conditions.

Case 2) All activity was released over one hour at tornado conditions. Case 2 resulted in slightly higher Control Room doses.

The TMA dose analysis assumptions are listed on Table 11.1. The activity released from the pool is listed on Table 11.5. The Control Room assumptions are listed on Table 11.2. The Control Room is assumed to be isolated within 60 seconds via the radiation monitors. A comparison of the nuclide concentration in the Control Room intake for the TMA to the radiation monitor response showed a Control Room isolation signal would occur before the 60 seconds assumed in the calculation. The resulting doses are presented on Table 11.4. Since the release from the TMA is assumed to end after one or two hours, the dose calculations are terminated after all contributions are accounted for; 2 hours for EAB and 24 hours for the Control Room.

TABLE 11.1 TMA DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Reactor power, MwT(including 2% uncertainty)	1550
Power Peaking Factor	1.75
Number of damaged fuel assemblies Hot Cold	 5 4
Fission product inventory in damaged assemblies after decay	Values calculated
Time after reactor shutdown hot assemblies cold assemblies	 100 hours 60 days
Fuel rod gap fractions I-131 other halogens Kr-85 other noble gases	 0.08 0.05 0.1 0.05
Iodine species above water elemental iodine organic iodine	 0.57 0.43
Pool DF elemental iodine organic iodide particulate Overall Pool DF	 500 1 ∞ 200
Exhaust flow rate, cfm 1-hour activity release 2 -hour activity release	 1.545E5 7.685E4
Iodine removal efficiency for all forms	0
Atmospheric dispersion, X/Q, sec/m^3 EA. Tornado conditions Normal conditions	 1.74E-6 4.8E-4

TABLE 11.1 TMA DOSE ANALYSIS ASSUMPTIONS	
Parameter	**Value**
Breathing rate, m^3/sec EA. 0-8 hr	 3.47E-4

TABLE 11.2 CONTROL ROOM PARAMETERS		
Parameter	**Value**	
Habitable volume, ft^3	36,211	
Normal Operating Mode make-up air flow rate, cfm	 2000+10%	
Accident Operating Mode Recirculating air iodine removal efficiency, % elemental methyl particulate flow rate, cfm Unfiltered in-leakage, cfm	 90 70 98 6000-10% 300	
Breathing rate, m^3/sec	3.47E-4	
Occupancy factor 0-24 hr 24-96 hr 96-720 hr	 1 0.6 0.4	
Atmospheric dispersion, X/Q, sec/m^3	1 hour release 4.36E-5 (Case 1)	2 hour release 4.36E-5 1.45E-3 (Case 2)

Table 11.3 **Flow Rate and Iodine Removal Schedule**				
Time, hours	Inleakage		Recirculation	
	cfm	iodine removal efficiency, %	cfm	iodine removal efficiency, %[1]
0-0.0167[2]	2200	0/0/0	0	0/0/0
0.0167-0.0194[3]	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	5400	90/70/98

TABLE 11.4 TMA DOSE, Rem, TEDE		
TMA	EAB, Max - 2 hours[4]	Control Room, 24 hours
1-hour release	2.01 E-2	5.87 E-2
2-hour release	7.41 E-2	5.44 E-2
Acceptance Criteria	6.3	5

[1]Elemental/Methyl/Particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

[4]The 2 hour LPZ dose is bounded by the 2-hour dose at the EAB, as such, only the EAB dose is evaluated.

TABLE 11.5
Spent Fuel Pool Activity

	A_{100}	A_{60d}	n	Xgap	Xpeak	DF	$A_{released}$
1-131	2.98E+07	2.432E+05	121	0.08	1.75	200	8.676E+02
1-132	2.52E+07	0.000E+00	121	0.05	1.75	200	4.557E+02
1-133	3.12E+06	1.261E-13	121	0.05	1.75	200	5.640E+01
1-134	0.00E+00	0.00E-0	121	0.05	1.75	200	0.0
1-135	2.23E+03	0.00	121	0.05	1.75	200	4.028E-02
Kr-85m	2.15E+00	0.00	121	0.05	1.75	1	7.774E-03
Kr-85	4.98E+05	4.934E+05	121	0.1	1.75	1	6.456E+03
Kr-87	4.58E-17	0.0	121	0.05	1.75	1	1.656E-19
Kr-88	7.48E-04	0.0	121	0.05	1.75	1	2.705E-06
Xe-131m	4.42E+05	3.084E+04	121	0.05	1.75	1	1.687E+03
Xe-133m	1.10E+06	2.416E-02	121	0.05	1.75	1	3.977E+03
Xe-133	5.71E+07	3.662E+04	121	0.05	1.75	1	2.066+05
Xe-135m	3.57E+02	0.0	121	0.05	1.75	1	1.291E+00
Xe-135	1.09E+05	0.0	121	0.05	1.75	1	3.941E+02
Xe-138	0.00E+00	0.0	121	0.05	1.75	1	0.0

Total core activity @ 100 hours (A_{100}): Ci

Total core activity @ 60 days (A_{60d}): Ci

Core assemblies (n)

Gap Fraction (Xgap)

Peaking factor (Xpeak)

Overall pool DF

Activity released from the pool to the environment ($A_{released}$):

$$A_{hot}:\ A_{hot} = \frac{A_{100}}{n} * 5$$

$$A_{cold}:\ A_{cold} = \frac{A_{60d}}{n} * 13$$

$$A_{total}:\ A_{total} = A_{hot} + A_{cold}$$

$$A_{total}: = \frac{A_{total} * X_{gap} * X_{peak}}{DF}$$

12.0 Waste Gas Decay Tank Rupture

12.1 Analysis

This analysis calculates the Control Room and off-site doses for a release of a Gas Decay Tank (GDT) into the Auxiliary Building Atmosphere

12.2 Assumptions

The source term is 100,000 Ci of equivalent Xe-133. The assumed source will be 100,000 Ci of actual Xe-133.

Activity, from the ruptured tank, is released to the environment over 2-hours. The flow rate, corresponding to a 2-hour activity release, is calculated in Section 7.1.6.

The 2-hour activity release assumption is consistent with that of the Fuel Handling Accident.

Activity from the ruptured tank is released into the Auxiliary Building and assumed to diffuse from the building to the environment. The Control Room dose calculation will use χ/Qs for the Auxiliary Building area source.

The dose conversion factor for Xe-133, contained in HABIT library MLWR1465.pwr, will be used. The DCF values in this library were derived from FGR 11 and 12.

Table 12.1
Atmospheric Dispersion (sec/m^3)

EAB	LPZ
4.8E-4	3.0E-5

Control Room

0-2 hours	2-8 hours	8-24 hours	24 - 96
3.89E-3	2.99E-3	9.63E-4	8.98E-4

Table 12.2
Control Room Parameters

Parameter	Value
Habitable volume, ft^3	36,211
Normal Operating Mode make-up air flow rate, cfm	 2000+10%
Accident Operating Mode This analysis considers only noble gas, as such, iodine removal efficiencies and recirculation flow have no effect on the calculated doses. Unfiltered in-leakage, cfm	300

Table 12.3
Flow Rate and Iodine Removal Schedule

Time, hours	Inleakage		Recirculation	
	cfm	iodine removal efficiency, % [1]	cfm	iodine removal efficiency, % [1]
0 - 0.0167[2]	2200	0/0/0	0	0/0/0
0.0167 - 0.0194[3]	300	0/0/0	0	0/0/0
>0.0194	300	0/0/0	0	0/0/0

Note: The isolation and recirculation times, shown above, are consistent with those provided for other accidents (excluding SGTR).

The iodine removal efficiencies and recirculation flow rates are not applicable to the GDT rupture, which assumes only Xe-133 in the source term (no iodine).

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[1]Elemental/Methyl/Particulate

[2]0 to 60 seconds

[3]60 to 70 seconds

Table 12.4
Offsite and Control Room Doses

	rem, TEDE		
	EAB Max. 2-hour	LPZ 2 hour	Control Room 24 hour
without CR isolation	2.77E-1	1.73E-2	6.63E-2
Acceptance Criteria	0.5	0.5	5
with CR isolation	2.77E-1	1.73E-2	9.56E-2
Acceptance Criteria	0.5	0.5	5

13.0 References

1. NUREG/CR-6331, Rev. 1 "Atmospheric Relative Concentrations in Building Wakes", J. V. Ramsdell, C. A. Simonen, Pacific Northwest National Laboratory, 1997

2. Draft Regulatory Guide DG-1111, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants, December 2001

3. Ginna UFSAR, Rev. 117, 10/02

4. NUREG - 1465, "Accident Source Terms for Light-Water Nuclear Power Plants", February 1995

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

6. HABIT Version 1.1, "Computer Codes for Evaluation of Control Room Habitability", TACT5 and CONHAB Modules, NUREG/CR-6210, Supplement 1

7. NUREG/CR-5966, "A simplified Model of Aerosol Removal by Containment Sprays", D. A. Powers, et al., Sandia National Laboratories, June 1993

8. NUREG/CR-6604, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal And Dose Estimation", S.L. Humphreys, et. al., Sandia National Laboratories, April 1998. (See Section 10)

9. NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations", T.J. Bander, USNRC, 1982

10. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion", Keith F. Eckerman, et al., Oak Ridge National Laboratory, 1988

11. Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil", Keith F. Eckerman, et al., Oak Ridge National Laboratory, 1993

12. A. G. Croff, "A User's Manual for the ORIGEN2 Computer Code", ORNL/TM-7175, Oak Ridge National Laboratory, July 1980

1. RG&E Design Analysis DA-NS-2001-060, Atmospheric Dispersion Factors for the Control Room Intake, Rev 0

2. RG&E Design Analysis DA-NS-2003-004, Atmospheric Dispersion Factors for the Exclusion Boundary and Low Population Zone, Rev 0

3. RG&E Design Analysis DA-NS-2001-063, Iodine and Noble Gas Activity in the Primary Coolant and Iodine Activity in the Secondary Coolant, Rev 1

4. RG&E Design Analysis DA-NS-2001-064, Iodine Appearance Rates, Rev 1

5. RG&E Design Analysis DA-NS-2001-087, Large Break LOCA Offsite And Control Room Doses, Rev 1

6. RG&E Design Analysis DA-NS-2002-004, Fuel Handling Accident Offsite and Control Room Doses, Rev 1

7. RG&E Design Analysis DA-NS-2002-007, Main Steam Line Break Offsite and Control Room Doses, Rev 2

8. RG&E Design Analysis DA-NS-2001-084, Steam Generator Tube Rupture Offsite and Control Room Doses, Rev 1

9. RG&E Design Analysis DA-NS-2002-054, Locked Rotor Offsite and Control Room Doses, Rev 0

10. RG&E Design Analysis DA-NS-2002-050, Control Rod Ejection Accident Offsite and Control Room Doses, Rev 0

11. RG&E Design Analysis DA-NS-2002-019, Tornado Missile Accident Offsite and Control Room Doses, Rev 1

12. RG&E Design Analysis DA-NS-2000-057, Gas Decay Tank Rupture Offsite and Control Room Doses, Rev 1

13. RG&E Design Analysis DA-NS-2002-037, HABIT Code Nuclear Data Library, Rev 0

Attachment 2

Summary of Control Room Toxic Hazards Analysis

Toxic Gas Evaluation Summary
CREATS/AST Submittal

Ginna Station Design Analysis DA-NS-2000-053, Control Room Toxic Gas Hazards Analysis, was performed to evaluate toxins that could threaten Control Room Habitability and, where appropriate, calculate the worst case substance concentrations within the Control Room. A Control Room in-leakage of 300 CFM is assumed after isolation, consistent with the radioactive dose analysis assumption.

The following toxins were evaluated:

- Ammonia
- Chlorine
- Sodium Hypochlorite
- Halon
- Refrigerant R-22
- Carbon Dioxide
- Other Miscellaneous chemicals

Summary of Analyses & Conclusions:

1.0 HABIT Computer analyses

Peak ammonia, chlorine, and sodium hypochlorite concentrations were calculated using the EXTRAN and CHEM codes from the HABIT package that is described in NUREG/CR-6210, PNL-10496; "Computer Codes for Evaluation of Control Room Habitability". All three calculations used the following inputs:

- Storage temperature: assumed ambient and conservatively high; 100°F / 40°C
- Ground Temperature: conservatively assumed same as air, 100°F / 40°C
- Air Temperature: 100°F / 40°C
- Wind speed: 1m / second
- Atmospheric Stability Class: F
- Atmospheric Pressure: 14.7 psia / 760 mmHg
- Solar Radiation: 1050 w/m^2
- Cloud Cover: 10 tenths Assumed value. Increased cloud cover maximizes the solar flux. The maximum value for cloud cover is 10.
- Release height = 0 m (ground puddle assumed)
- Intake height = 5.8m
- Unfiltered outside air is assumed to flow into the Control Room at the maximum rate of 2200 CFM for 30 seconds before Control Room Emergency Zone (CREZ) isolation occurs. Either of two redundant chlorine monitors in the outside air intake will isolate the CREZ and reduce the unfiltered air inflow to a maximum of 300 CFM for the duration of the event.

1.1 Ammonia analysis using HABIT

The 4000 gallon ammonium hydroxide tank is located outside, on the north side of the Turbine Building, at ground elevation 253'. The Control Building (and it's outside air intake) is located 200' away on the south side of the Turbine building. The outside air intake is located at elevation 315' and the roof line of the Turbine building, which intervenes between the source and the intake, is at elevation of 361'. In addition to the HABIT inputs described above, inputs used specifically for the ammonia analysis are as follows:

- Initial mass of ammonium hydroxide = 13,600 kg, S.G. = 0.9
- Maximum pool radius = 2.74m
- Distance to intake = 60m
- Building area (for most conservative wake effect)= 24m^2

Results: The peak ammonia concentration in the Control Room was 31.9 g/m^3, which is less than the 210 mg/m^3 limit found in Regulatory Guide 1.78, Rev. 1, December 2001. The 210-mg/m^3 limit would be reached if unfiltered in-leakage to the Control Room increased to 16,000 CFM, which greatly exceeds the total outside airflow into the Control Room (normally ≤2000 CFM).

1.2 Chlorine analysis using HABIT

Approximately 1.1 miles east of Ginna Station is a water treatment plant that uses two 2000 lb tanks of liquefied chlorine to treat lake water for distribution through the Ontario water system. The analysis assumes catastrophic failure of 1 tank. In addition to the HABIT inputs described above, inputs used specifically for the chlorine analysis are as follows:

- Initial mass of chlorine = 908 kg
- Distance to intake = 1770m
- The analysis assumes catastrophic failure of 1 tank.

Results: The peak chlorine concentration in the Control Room was 18.2 mg/m^3, which is less than the 30 mg/m^3 limit found in Regulatory Guide 1.78, Rev. 1, dated December 2001. The 30-mg/m^3 limit would be reached if unfiltered in-leakage to the Control Room increased to 500 CFM.

1.3 Sodium Hypochlorite analysis using HABIT

Approximately 310' due north of the Control Room is a non-pressurized plastic tank containing 2500 gallons of 16% sodium hypochlorite (NaOCl). The tank is at ground elevation 253', and the Control Room outside air intake is at elevation 313'. The Turbine Building intervenes between the tank and the intake. It is immediately to the north of the outside air intake, with its roof at elevation 360'. The NaOCl tank is inside of a16' square concrete dike that would limit the size of a spill or rupture. In addition to the HABIT inputs described above, inputs used specifically for the NaOCL analysis are as follows:

- Initial mass of sodium hypochlorite = 11,500 kg, S.G. = 1.218
- Maximum pool radius = 2.74m
- Distance to intake = 94.5 m
- Building area (for most conservative wake effect)= $24m^2$

Results: Sodium Hypochlorite located onsite at Ginna is located far from the Control Room and is quite stable. The worst case calculated concentration of sodium hypochlorite outside of the Control Room was less than 0.0004 mg/m^3 and thus considered a negligible threat to Control Room habitability.

2.0 Halon toxin assessment

Halon is used for fire suppression in the Relay Room located one level below the Control Room. There are no other sources of Halon that could threaten Control Room habitability. At one time the Relay Room contained a smaller MUX room which was protected by a separate halon system; discharge testing of these two halon systems yielded concentrations of 6.4% in the Relay Room and 8.5% in the MUX room (volume concentrations, measured at three locations in each room).

These rooms and their halon systems have since been combined into a single zone whose total volume is almost identical to the CREZ volume. In the worst case scenario of thoroughly mixing the entire halon inventory between the Relay and Control Rooms, the 6.4% and 8.4% values would be cut in half. The diluted concentration would be less than the 5% exposure limit recommended in NUREG/CR-5669; "Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators"

Results: The Relay Room halon system is not considered a threat to CREZ habitability, and there are no other sources of halon that could threaten Control Room habitability. The worst case scenario does not exceed acceptable limits, and that scenario is conservative because the Relay Room and CREZ HVAC systems are not directly connected, and because two normally closed doors separate the CREZ from the Relay Room. Thus, it would be highly unlikely for the entire halon inventory to thoroughly mix between the Relay and Control Rooms.

3.0 Refrigerant R-22 toxin assessment

RG&E established a conservative exposure limit of 3000 PPM for not more than 30 minutes, and never to exceed 5000 PPM. This limit is more conservative than the 42,000 PPM limit found in ANSI/ASHRAE Std. 15, and was proposed by RG&E to NRC staff in a meeting held 2/28/2001 at NRC offices in Rockford, MD. As documented in the NRC's summary letter dated 3/21/2001, this limit was considered conservative and acceptable.

The analysis conservatively assumes that a single coil ruptures and immediately releases that cooling circuit's entire inventory into the CREZ.

CREZ volume is 36,211 ft^3.

R-22 has a vapor density of 2.76 (air =1). Air at 70° F & 50% RH has a density of approximately 0.073 lb/ft^3, making the density of R-22 equal to: 0.073 X 2.76 = 0.20 lb/ft^3, and the specific volume of 5.0 ft^3/lb.

Results: Each new CREATS cooling system circuit will contain ≤ 29.0 lbs of R-22. This design limit will prevent exposure above limits of 3000 PPM for no more than 30 minutes, and never to exceed 5000 PPM.

4.0 Carbon dioxide assessment

CO2 was considered in accordance with Standard Review Plan section 6.4, paragraph III-2. The latest Revision 1 of Regulatory Guide 1.78 indicates an IDLH value of 4% by volume for CO2. RG&E used a more conservative 2% limit.

Air normally contains 0.0314% carbon dioxide; thus every ft^3 of air can absorb 0.0197 ft3 of CO2 before reaching the 2% limit.

The only source of CO2 in the CREZ is occupant respiration, which was conservatively assumed equal to 2.20 cubic foot/man-hour. Discharge of portable fire extinguisher(s) is not considered because accident analysis does not require postulating a fire in the Control Room concurrent with a radiological event that requires Control Room isolation.

Vapor density of CO2 is 1.53 (air =1). Air at 70°F & 50% RH has a density of approximately 0.073 lb/ft^3, making the density of CO2 equal to: 0.073 X 1.53 = 0.112 lb/ft^3.

Results: Carbon dioxide is not considered a threat to CREZ habitability because:

1) Assuming zero in-leakage, 320 man-hours would pass before CO2 levels rise to the 2% limit. A long-term uncontrolled release requiring a continuous CREZ isolation longer than 320 man-hours is a very low probability event.

2) In-leakage of less than 2.0 CFM/occupant will offset CO2 generation and preclude the buildup of CO2 to an unacceptable level. Ginna does not, by design or control, admit unfiltered air into the Control Room when isolated in the emergency mode of operation, but it is likely that at least 2.0 CFM /occupant leaks into the Control Room. This inleakage will offset CO2 generation and preclude the buildup of CO2 to an unacceptable level. In addition, RG&E has the capability to monitor CO2 levels in the long term, should it become necessary.

5.0 Assessment of Other chemicals

Other chemicals used onsite were inspected and are not considered a threat to Control Room habitability because of their volume, volatility, and/or their location relative to the Control Room's outside air intake. These chemicals include:

- Sulfuric acid at 95%; ≤6000 gallons in a non-pressurized tank that is located within a dike, inside of the AVT building, 100' north of the Control Building.
- Sodium hydroxide at 50%; ≤6000 gallons in a non-pressurized tank that is located within a dike, inside of the AVT building, 100' north of the Control Building.
- Sodium hydroxide at 30%, ≤5100 gallons in a tank located in the Auxiliary Building basement. That tank is equipped with a low pressure nitrogen blanket.
- Hydrazine at 35%, two 30 gallon drums located in the northeast corner of the Turbine Building, middle level. Drums are kept in a secondary container and/or berm that will limit the spread of a spill.
- ETA (Monoethanolamine CAS # 141-43-5) at 40%, two 350 gallon tanks located in the northeast corner of the Turbine Building basement. Each tank has a low pressure nitrogen blanket and bermed tank holders, which would contain any leakage within the stand on which the tank rests.

6.0 Smoke assessment

Smoke and fire were considered and their impact upon Control Room habitability is described in section 6.4.4.3 of the draft UFSAR description (attachment #8 of the License Amendment Request).

Attachment 3

Proposed Technical Specification Change Markup

Section 1.1, Definitions

Section 3.3.6, Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

Section 3.4.16, RCS Specific Activity

Section 3.6.6, Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC), NaOH, and Containment Post Accident Charcoal Systems

Section 3.7.9, Control Room Emergency Air Treatment System (CREATS)

Section 5.5.10, Ventilation Filter Testing Program (VFTP)

Section 5.5.16, Control Room Integrity Program

Section 5.6.7, Control Room Emergency Filtration System Report

1.0 USE AND APPLICATION

1.1 Definitions

- NOTE -

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
ACTUATION LOGIC TEST	An ACTUATION LOGIC TEST shall be the application of various simulated or actual input combinations in conjunction with each possible interlock logic state and the verification of the required logic output. The ACTUATION LOGIC TEST, as a minimum, shall include a continuity check of output devices.
AXIAL FLUX DIFFERENCE (AFD)	AFD shall be the difference in normalized flux signals between the top and bottom halves of a two section excore neutron detector.
CHANNEL CALIBRATION	A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, interlock, display, and trip functions. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.
CHANNEL CHECK	A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

CHANNEL OPERATIONAL TEST (COT)	A COT shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the OPERABILITY of required alarm, interlock, display, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.
CORE ALTERATIONS	CORE ALTERATIONS 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. 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 plant specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter 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.
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 E-7 of Regulatory Guide 1.109, Revision 1, 1977.~~
$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY	$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies (in MeV) per disintegration for non-iodine isotopes, with half lives > 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ICRP 30, Supplement to Part 1, page 192-212, table entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"

LEAKAGE

LEAKAGE from the RCS shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE - MODES

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, Initial Test Program of the UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission (NRC).

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The PTLR is the plant specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the power operated relief valve lift settings and enable temperature associated with the Low Temperature Overpressurization Protection System for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these limits is addressed in individual specifications.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the highest average nuclear power in any quadrant to the average nuclear power in the four quadrants.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 1520 MWt.

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCAs not capable of being fully inserted, the reactivity worth of the RCCAs must be accounted for in the determination of SDM; and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal hot zero power temperature.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT)	A TADOT shall consist of operating the trip actuating device and verifying the OPERABILITY of required alarm, interlock, display, and trip functions. The TADOT shall include adjustment, as necessary, of the trip actuating device so that it actuates at the required setpoint within the required accuracy.

Table 1.1-1
MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	% RATED THERMAL POWER[a]	AVERAGE REACTOR COOLANT TEMPERATURE (°F)
1	Power Operation	≥ 0.99	> 5	NA
2	Startup	≥0.99	≤ 5	NA
3	Hot Shutdown	< 0.99	NA	≥ 350
4	Hot Standby[b]	< 0.99	NA	350 > T_{avg} > 200
5	Cold Shutdown[b]	< 0.99	NA	≤ 200
6	Refueling[c]	NA	NA	NA

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

3.3 INSTRUMENTATION

3.3.6 Control Room Emergency Air Treatment System (CREATS) Actuation Instrumentation

LCO 3.3.6 The CREATS actuation instrumentation for each Function in Table 3.3.6-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
~~During CORE ALTERATIONS.~~

ACTIONS

- NOTE -
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel inoperable.	A.1 ~~- NOTE - The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition.~~ Place CREATS in ~~Mode F.~~ emergency mode	~~1~~ Y hour
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies ~~or during CORE ALTERATIONS.~~	~~C.1 Suspend CORE ALTERATIONS.~~ ~~AND~~ C.~~2~~ 1 Suspend movement of irradiated fuel assemblies.	~~Immediately~~ Immediately

SURVEILLANCE REQUIREMENTS

- NOTE -

Refer to Table 3.3.6-1 to determine which SRs apply for each CREATS Actuation Function.

	SURVEILLANCE	FREQUENCY
SR 3.3.6.1	Perform COT.	92 days
SR 3.3.6.2	- NOTE - Verification of setpoint is not required. Perform TADOT.	24 months
SR 3.3.6.3	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.6.4	Perform ACTUATION LOGIC TEST.	24 months

Table 3.3.6-1
CREATS Actuation Instrumentation

	FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1.	Manual Initiation	1 train	SR 3.3.6.2	NA
2.	Automatic Actuation Logic and Actuation Relays	1 train	SR 3.3.6.4	NA
3.	Control Room Radiation Intake Monitor			
	a. Iodine	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 9 \times 10^{-9}$ µCi/cc
	b. Noble Gas	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 1 \times 10^{-5}$ µCi/cc
	c. Particulate	1	SR 3.3.6.1 SR 3.3.6.3	$\leq 1 \times 10^{-8}$ µCi/cc
4.	Safety Injection	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 1, for all initiation functions and requirements.		

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.16 RCS Specific Activity

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

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. DOSE EQUIVALENT I-131 specific activity not within limit.	- NOTE - LCO 3.0.4 is not applicable.	
	A.1 Verify DOSE EQUIVALENT I-131 ~~within the acceptable region of Figure 3.4.16-1.~~ ≤ 60 µCi/gm	Once per 8 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limit.	7 days
B. Required Action and associated Completion Time of Condition A not met. OR DOSE EQUIVALENT I-131 specific activity ~~in the unacceptable region of Figure 3.4.16-1.~~ >60 µCi/gm	B.1 Be in MODE 3 with T_{avg} < 500°F.	8 hours
C. Gross specific activity not within limit.	C.1 Be in MODE 3 with T_{avg} < 500°F.	8 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.16.1	Verify reactor coolant gross specific activity ≤ $100/\bar{E}$ μCi/gm.	7 days
SR 3.4.16.2	- NOTE - Only required to be performed in MODE 1. Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 1.0 μCi/gm.	14 days <u>AND</u> Between 2 and 10 hours after a THERMAL POWER change of ≥ 15% RTP within a 1 hour period
SR 3.4.16.3	- NOTE - Only required to be performed in MODE 1. Determine $\bar{E}$ from a reactor coolant sample.	Once within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. <u>AND</u> Every 184 days thereafter

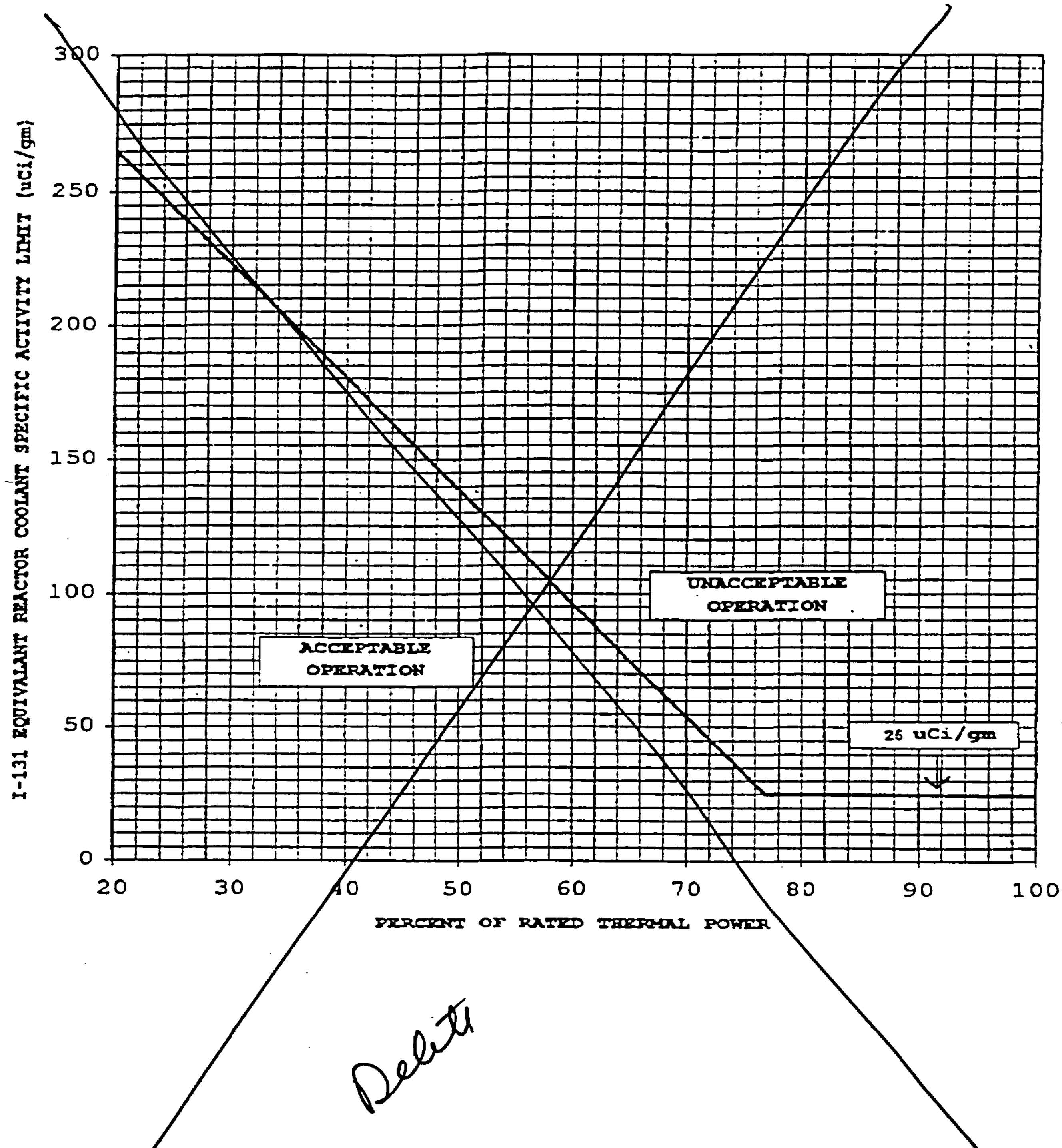

Figure 3.4.16-1
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity Limit Versus Percent of RATED THERMAL POWER

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray (CS), Containment Recirculation Fan Cooler (CRFC)~~,~~ and NaOH, ~~and Containment Post-Accident Charcoal Systems~~ Systems

LCO 3.6.6 Two CS trains, four CRFC units, ~~two post-accident charcoal filter trains~~, and the NaOH system shall be OPERABLE.

- NOTE -

In MODE 4, both CS pumps may be in pull-stop for up to 2 hours for the performance of interlock and valve testing of motor operated valves (MOVs) 857A, 857B, and 857C. Power may also be restored to MOVs 896A and 896B, and the valves placed in the closed position, for up to 2 hours for the purpose of each test.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One CS train inoperable.	A.1	Restore CS train to OPERABLE status.	72 hours
~~B.~~	~~One post-accident charcoal filter train inoperable.~~	~~B.1~~	~~Restore post-accident charcoal filter to OPERABLE status.~~	~~7 days~~
~~C.~~	~~Two post-accident charcoal filter trains inoperable.~~	~~C.1~~	~~Restore one post-accident charcoal filter train to OPERABLE status.~~	~~72 hours~~
B ~~D~~.	NaOH system inoperable.	B ~~D~~.1	Restore NaOH System to OPERABLE status.	72 hours
C ~~E~~.	Required Action and associated Completion Time of Condition A or B~~, C, or D~~ not met.	C ~~E~~.1	Be in MODE 3.	6 hours
		AND		
		C ~~E~~.2	Be in MODE 5.	84 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D ~~F~~. One or two CRFC units inoperable.	~~F.1~~	~~- NOTE - Required Action F.1 only required if CRFC unit A or C is inoperable.~~ ~~Declare associated post-accident charcoal filter train inoperable.~~	~~Immediately~~
	~~AND~~		
	D.1 ~~F.2~~	Restore CRFC unit(s) to OPERABLE status.	7 days
E ~~G~~. Required Action and associated Completion Time of Condition D ~~F~~ not met.	E ~~G~~.1	Be in MODE 3.	6 hours
	AND		
	E ~~G~~.2	Be in MODE 5.	36 hours
F ~~H~~. Two CS trains inoperable. ~~OR~~ ~~NaOH System and one or both post-accident charcoal filter trains inoperable.~~ OR ~~OR~~ Three or more CRFC units inoperable. ~~OR~~ ~~One CS and two post-accident charcoal filter trains inoperable.~~	F ~~H~~.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.6.1	Perform SR 3.5.2.1 and SR 3.5.2.3 for valves 896A and 896B.	In accordance with applicable SRs.
SR 3.6.6.2	Verify each CS manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.3	Verify each NaOH System manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position.	31 days
SR 3.6.6.4	Operate each CRFC unit for ≥ 15 minutes.	31 days
SR 3.6.6.5	Verify cooling water flow through each CRFC unit.	31 days
~~SR 3.6.6.6~~	~~Operate each post-accident charcoal filter train for ≥ 15 minutes.~~	~~31 days~~
SR 3.6.6.~~7~~6	Verify each CS pump's developed head at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.6.6.~~8~~7	Verify NaOH System solution volume is ≥ ~~4500~~ 3000 gal.	184 days
SR 3.6.6.~~9~~8	Verify NaOH System tank NaOH solution concentration is ≥ 30% and ≤ 35% by weight.	184 days
~~SR 3.6.6.10~~	~~Perform required post-accident charcoal filter testing in accordance with the Ventilation Filter Testing Program (VFTP).~~	~~In accordance with the VFTP~~
SR 3.6.6.~~11~~9	Perform required CRFC unit testing in accordance with the VFTP.	In accordance with the VFTP
SR 3.6.6.~~12~~10	Verify each automatic CS valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.~~13~~11	Verify each CS pump starts automatically on an actual or simulated actuation signal.	24 months

SURVEILLANCE		FREQUENCY
SR 3.6.6.~~14~~ 12	Verify each CRFC unit starts automatically on an actual or simulated actuation signal.	24 months
~~SR 3.6.6.15~~	~~Verify each post-accident charcoal filter train damper actuates on an actual or simulated actuation signal.~~	~~24 months~~
SR 3.6.6.~~16~~ 13	Verify each automatic NaOH System valve in the flow path that is not locked, sealed, or otherwise secured in position actuates to the correct position on an actual or simulated actuation signal.	24 months
SR 3.6.6.~~17~~ 14	Verify spray additive flow through each eductor path.	5 years
SR 3.6.6.~~18~~ 15	Verify each spray nozzle is unobstructed.	10 years

3.7 PLANT SYSTEMS

3.7.9 Control Room Emergency Air Treatment System (CREATS)

LCO 3.7.9 The CREATS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
During movement of irradiated fuel assemblies,
During CORE ALTERATIONS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CREATS filtration train inoperable.	A.1 Restore CREATS filtration train to OPERABLE status.	48 hours
	<u>OR</u>	
	A.2 - NOTE - The control room may be unisolated for ≤ 1 hour every 24 hours while in this condition.	
	Place isolation dampers in CREATS Mode F.	48 hours
B. - NOTE - Separate Condition entry allowed for each damper. One CREATS isolation damper in one or more outside air flowpaths inoperable.	B.1 Restore isolation damper to OPERABLE status.	7 days

Replace with attached

CONDITION		REQUIRED ACTION		COMPLETION TIME
C.	Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, 3, or 4.	C.1	Be in MODE 3.	6 hours
		<u>AND</u>		
		C.2	Be in MODE 5.	36 hours
D.	Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel or during CORE ALTERATIONS.	D.1	Place OPERABLE isolation damper(s) in CREATS Mode F.	Immediately
		<u>OR</u>		
		D.2.1	Suspend CORE ALTERATIONS.	Immediately
			<u>AND</u>	
		D.2.2	Suspend movement of irradiated fuel assemblies.	Immediately
E.	Two CREATS isolation dampers for one or more outside air flow paths inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately
F.	Two CREATS isolation dampers for one or more outside air flow paths inoperable during movement of irradiated fuel assemblies or during CORE ALTERATIONS.	F.1	Suspend CORE ALTERATIONS.	Immediately
		<u>AND</u>		
		F.2	Suspend movement of irradiated fuel assemblies.	Immediately

Replace with attached

SURVEILLANCE REQUIREMENTS

Replace with attached

SURVEILLANCE		FREQUENCY
SR 3.7.9.1	Operate the CREATS filtration train ≥ 15 minutes.	31 days
SR 3.7.9.2	Perform required CREATS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with VFTP
SR 3.7.9.3	Verify the CREATS actuates on an actual or simulated actuation signal.	24 months

3.7 PLANT SYSTEMS

3.7.~~10~~ 9 Control Room Emergency ~~Filtration~~ Air Treatment System ~~(CREFS)~~ CREATS

LCO 3.7.~~10~~ 9 Two ~~CREFS~~ CREATS trains and the control room boundary shall be OPERABLE.

- NOTE -
The control room boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, 4, ~~[5, and 6]~~,
During movement of ~~[recently]~~ irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~CREFS~~ CREATS train inoperable.	A.1 Restore ~~CREFS~~ CREATS train to OPERABLE status.	7 days
B. ~~Two CREFS trains inoperable due to inoperable~~ control room boundary inoperable ~~in MODE 1, 2, 3, or 4.~~	B.1 Initiate compensatory measures AND B.~~1~~ 2 Restore control room boundary to OPERABLE status.	Immediately ~~24 hours~~ 14 days
C. Required Action and Associated Completion Time of Condition B not met	C.1 Initiate action in accordance with Specification 5.6.7.	Immediately
~~C~~ D. Required Action and associated Completion Time of Condition A ~~or B~~ not met in MODE 1, 2, 3, or 4.	~~C~~ D.1 Be in MODE 3. AND ~~C~~ D.2 Be in MODE 5.	6 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D.~~ E Required Action and associated Completion Time of Condition A not met ~~[in MODE 5 or 6, or]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	~~D.1~~ E.1 ~~- NOTE - [Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable.]~~ Place OPERABLE CREATS ~~CREFS~~ train in emergency mode.	Immediately
	OR	
	~~D.2~~ E.2 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
~~E.~~ F Two CREATS ~~CREFS~~ trains inoperable ~~[in MODE 5 or 6, or]~~ during movement of ~~[recently]~~ irradiated fuel assemblies.	~~E.1~~ F.1 Suspend movement of ~~[recently]~~ irradiated fuel assemblies.	Immediately
~~F.~~ G Two CREATS ~~CREFS~~ trains inoperable in MODE 1, 2, 3, or 4 ~~for reasons other than Condition B~~.	~~F.1~~ G.1 Enter LCO 3.0.3	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.~~10~~9.1	Operate each CREATS ~~CREFS~~ train for ~~[≥ 10 continuous hours with the heaters operating or (for systems without heaters)~~ ≥ 15 minutes~~]~~.	31 days
SR 3.7.~~10~~9.2	Perform required CREATS ~~CREFS~~ filter testing in accordance with the ~~[~~Ventilation Filter Testing Program (VFTP)~~]~~.	In accordance with the ~~[~~VFTP~~]~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.~~10~~ 9.3	Verify each ~~CREFS~~ CREATS train actuates on an actual or simulated actuation signal.	~~[18]~~ 24 months
SR 3.7.~~10~~ 9.4	~~Verify one CREFS train can maintain a positive pressure of ≥ [0.125] inches water gauge, relative to the adjacent [turbine building] during the pressurization mode of operation at a makeup flow rate of ≤ [3000] cfm.~~ Verify control room habitability requirements are met in accordance with the Control Room Integrity Program (CRIP)	~~[18] months on a STAGGERED TEST BASIS~~ In accordance with the CRIP

5.0 ADMINISTRATIVE CONTROLS

5.5 Programs and Manuals

The following programs and manuals shall be established, implemented, and maintained.

5.5.1 <u>Offsite Dose Calculation Manual (ODCM)</u>

The ODCM shall contain:

a. The methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program; and

b. The radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s),

   2. a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and does not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after review and acceptance by the onsite review function and the approval of the plant manager; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment Program

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident. The systems include Containment Spray, Safety Injection, and Residual Heat Removal in the recirculation configuration. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

5.5.3 Deleted

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the plant to unrestricted areas, conforming to 10 CFR 50, Appendix I and 40 CFR 141;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table 2, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the plant to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

5.5.5 Component Cyclic or Transient Limit Program

This program provides controls to track the reactor coolant system cyclic and transient occurrences specified in UFSAR Table 5.1-4 to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 2.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8 Steam Generator (SG) Tube Surveillance Program

Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g). The program shall include the following:

a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program.

b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.

c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.

5.5.9 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

~~a.~~ Containment Post-Accident Charcoal System

1. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

~~3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.~~

a ~~b~~. Containment Recirculation Fan Cooler System

1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

b ~~c~~. Control Room Emergency Air Treatment System (CREATS)

1. Demonstrate the pressure drop across the Combined HEPA filters, the prefilters, the charcoal adsorbers, and the post filters ~~HEPA filter bank~~ is < ~~3~~ 14 inches of water at a design flow rate (± 10%).

2. Demonstrate that an in-place DOP test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

~~3. Demonstrate the pressure drop across the charcoal adsorber bank is < 3 inches of water at a design flow rate (± 10%).~~

3 ~~4~~. Demonstrate that an in-place Freon test of the charcoal adsorber bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

4 ~~5~~. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

c ~~d~~. SFP Charcoal Adsorber System

1. Demonstrate that the total air flow rate from the charcoal adsorbers shows at least 75% of that measured with a complete set of new adsorbers.

2. Demonstrate that an in-place Freon test of the charcoal adsorbers bank shows a penetration and system bypass < 1.0%, when tested under ambient conditions.

3. Demonstrate that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows a methyl iodide penetration of less than 14.5% when tested in accordance with ASTM D3803-1989 at a test temperature of 30°C (86°F) and a relative humidity of 95%.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP frequencies.

5.5.11 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas decay tanks and the quantity of radioactivity contained in waste gas decay tanks. The gaseous radioactivity quantities shall be determined following the methodology in NUREG-0133.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas decay tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

b. A surveillance program to ensure that the quantity of radioactivity contained in each waste gas decay tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color; and

b. Within 31 days following addition of the new fuel to the storage tanks, verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license; or

2. A change to the UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.13.b.1 or Specification 5.5.13.b.2 shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71e.

5.5.14 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the supported system(s) is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the inoperable support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.15 <u>Containment Leakage Rate Testing Program</u>

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 60 psig.

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.2% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. For each air lock, overall leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$, and

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at $\geq P_a$.

c. Mini-purge valve acceptance criteria is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

5.5.~~18~~ 16 Control Room Integrity Program (CRIP)

A ~~Control Room Integrity~~ Program shall be established and implemented to ensure that control room envelope integrity is maintained. The program shall provide controls to limit radioactive gas and toxic gas leakage into the control room from sources external to the control room envelope to levels that support control room habitability. The program shall include guidance on the following elements:

a. Defining the control room envelope boundaries;

b. Assessing control room habitability;

c. Testing for control room in-leakage; and

d. Maintaining control room envelope integrity.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures > 100 mrem/yr and their associated man rem exposure according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions. The report shall be submitted on or before April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the plant during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring activities for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the plant shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

The following administrative requirements apply to the COLR:

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

LCO 3.1.1,	"SHUTDOWN MARGIN (SDM)";
LCO 3.1.3,	"MODERATOR TEMPERATURE COEFFICIENT (MTC)";
LCO 3.1.5,	"Shutdown Bank Insertion Limit";
LCO 3.1.6,	"Control Bank Insertion Limits";
LCO 3.2.1,	"Heat Flux Hot Channel Factor ($F_Q(Z)$)";
LCO 3.2.2,	"Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)";
LCO 3.2.3,	"AXIAL FLUX DIFFERENCE (AFD)";
LCO 3.4.1,	"RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; and
LCO 3.9.1,	"Boron Concentration."

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.
   (Methodology for LCO 3.1.1, LCO 3.1.3, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.9.1.)

2. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: WCOBRA/TRAC Two-Loop Upper Plenum Injection Model Updates to Support ZIRLO™ Cladding Option," February 1994.
   (Methodology for LCO 3.2.1.)

3. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974.
   (Methodology for LCO 3.2.3.)

4. WCAP-12610-P-A, "VANTAGE + Fuel Assembly Reference Core Report," April 1995.
   (Methodology for LCO 3.2.1.)

5. WCAP 11397-P-A, "Revised Thermal Design Procedure," April 1989.
   (Methodology for LCO 3.4.1 when using RTDP.)

6. WCAP-10054-P-A and WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.
   (Methodology for LCO 3.2.1.)

7. WCAP-10924-P-A, Volume 1, Revision 1, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation Responses to NRC Questions," and Addenda 1,2,3, December 1988.
   (Methodology for LCO 3.2.1.)

8. WCAP-10924-P-A, Volume 2, Revision 2, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 2: Application to Two-Loop PWRs Equipped with Upper Plenum Injection," and Addendum 1, December 1988.
   (Methodology for LCO 3.2.1.)

9. WCAP-10924-P-A, Volume 1, Revision 1, Addendum 4, "Westinghouse Large-Break LOCA Best-Estimate Methodology, Volume 1: Model Description and Validation, Addendum 4: Model Revisions," March 1991.
   (Methodology for LCO 3.2.1.)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

The following administrative requirements apply to the PTLR:

a. RCS pressure and temperature limits for heatup, cooldown, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"

b. The power operated relief valve lift settings required to support the Low Temperature Overpressure Protection (LTOP) System, and the LTOP enable temperature shall be established and documented in the PTLR for the following:

LCO 3.4.6, "RCS Loops - MODE 4";

LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";

LCO 3.4.10, "Pressurizer Safety Valves"; and

LCO 3.4.12, "LTOP System."

c. The analytical methods used to determine the RCS pressure and temperature and LTOP limits shall be those previously reviewed and approved by the NRC in NRC letter, "R.E. Ginna - Acceptance for Referencing of Pressure Temperature Limits Report, Revision 2 (TAC No. M96529)," dated November 28, 1997. Specifically, the methodology is described in the following documents:

1. Letter from R.C. Mecredy, Rochester Gas and Electric Corporation (RG&E), to Document Control Desk, NRC, Attention: Guy S. Vissing, "Application for Facility Operating License, Revision to Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) Administrative Controls Requirements," Attachment VI,

September 29, 1997, as supplemented by letter from R.C. Mecredy, RG&E, to Guy S. Vissing, NRC, "Corrections to Proposed Low Temperature Overpressure Protection System Technical Specification," October 8, 1997.

2. WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Sections 1 and 2, January, 1996.

d. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for revisions or supplement thereto.

5.6.~~10~~ 7 Control Room Emergency ~~Filtration~~ Air Treatment System (CREATS Report

When a report is required by Condition C of LCO 3.7.~~10~~ 9, "Control Room Emergency ~~Filtration System (CREFS)~~ Air Treatment System (CREATS)," a report shall be submitted within the following 90 days. The report shall outline the compensatory measures, the cause of the inoperability, and the plans and schedule for restoring the ~~CREFS~~ CREATS to OPERABLE status.