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PSEG
Nuclear LLC

MAY 17 2001
LRN-01-152
LCR H01-002

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Gentlemen:

REQUEST FOR CHANGE TO TECHNICAL SPECIFICATIONS
INCREASE IN ALLOWABLE MSIV LEAKAGE RATE AND
ELIMINATION OF MSIV SEALING SYSTEM
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

Pursuant to 10CFR50.90, PSEG Nuclear LLC hereby requests a revision to the Technical Specifications (TS) for the Hope Creek Generating Station (HCGS). This amendment requests an increase in the allowable Main Steam Isolation Valve Leakage from 46 standard cubic feet per hour (scfh) to 250 scfh and deletion of the MSIV Sealing System (MSIVSS). The amendment also resolves the control room unfiltered inleakage issue by increasing the design value from 10 cubic feet per minute (cfm) to 900 cfm. In addition, this amendment requests full implementation of the alternate source term in accordance with 10CFR50.67. Based upon our review of the standards set forth in 10CFR50.92(c) we have concluded that the proposed amendment presents no significant hazards consideration.

Also, in accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

PSEG currently plans to implement the proposed changes by the end of the upcoming refueling outage scheduled for October 2001. Therefore PSEG requests that the NRC approve this proposed change by the end of September in order to support our scheduled implementation date of November 2, 2001.

An evaluation of the requested changes is provided in Attachment 2 to this letter. The marked up Technical Specification pages affected by the proposed changes are provided in Attachment 3. The supporting calculations are provided in Attachments 4 and 5.

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The following analysis is based on preliminary plant specific assumptions for core inventory fission products. A plant specific calculation is currently being finalized to verify pH levels above 7 and tracer gas testing will be performed to ensure that a control room in leakage of 900 cfm is bounding. Should any of the parameters above change during verification and become more limiting, we will revise our submittal accordingly.

Additional analysis is required to address NUREG-0737 such as shielding, access, sampling and affect on other units. This information will be available prior to implementation.

If you have any questions or require additional information, please contact Mr. Michael Mosier at (856) 339-5434.

Sincerely,

D. Garchow
Vice President – Operations

Attachments:
1. Notarized Affidavit
2. Licensee's Evaluation of revisions to the Technical Specifications (TS) for increased Main Steam Isolation Valve (MSIV) leakage and deletion of the Main Steam Isolation Valve Sealing System (MSIVSS)
3. Markup of Technical Specification and Bases pages
4. HCCALC H-1-ZZ-MDC-1880, Rev 0, Post-LOCA EAB, LPZ, and CR Doses – Alternate Source Term Analysis.
5. HCCALC H-1-ZZ-MDC-1879, Rev 0, Control Room Chi/Qs For FRVS Vent, RB Truck Bay, TB Louvers, and SPV.

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Manager - Hope Creek Operations (H01)
Manager – Nuclear Reliability Program (X07)
Manager – System Engineering – Hope Creek (H18)
Manager – Nuclear Safety and Licensing (N21)
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NBU RM (N64)
Microfilm Copy
File Number 1.2.1 (Hope Creek), 2.3 (LCR H01-002)

STATE OF NEW JERSEY)
) SS
COUNTY OF SALEM)

D. Garchow, being duly sworn according to law deposes and says:

I am Vice President – Operations of PSEG Nuclear LLC, and as such, I find the matters set forth in the above referenced letter, concerning Hope Creek Generating Station, Unit 1, are true to the best of my knowledge, information and belief.

David F. Garchow

Subscribed and Sworn to before me
this 16 day of May, 2001

Sheri L. Huston
Notary Public of New Jersey

My Commission expires on ______________.

SHERI L. HUSTON
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 12/08/2003

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

EVALUATION OF REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS) FOR INCREASED MSIV LEAKAGE AND DELETION OF THE MAIN STEAM ISOLATION VALVE SEALING SYSTEM (MSIVSS)

May 16, 2001

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1.0 INTRODUCTION

This letter is a request to amend Facility Operating License NPF-57 for the Hope Creek Generating Station, Unit 1. The proposed changes would revise the Technical Specifications contained in Appendix A to the Operating License to permit an increase in the allowable leak rate for the Main Steam Isolation Valves (MSIVs) and to delete the MSIVSS. These changes are based on the use of an alternative source term and the guidance provided in Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors". These changes will result in a reduction in occupational exposure from the reduction in MSIV-related maintenance work. The proposed changes will also provide an economic benefit by eliminating the high maintenance and operational expense associated with the MSIVSS.

Approval of these proposed changes is being requested by the end of September 2001 to support the scheduled implementation date of November 2, 2001.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed changes to the Technical Specifications are included in Attachment 3 of this submittal. In summary, it is requested that:

1. Allowable leak rate specified in TS 3.6.1.2 be changed from 46.0 total standard cubic feet per hour (scfh) to 150 scfh per main steam line, as well as 250 scfh leak rate combined for all four steam lines. The proposed change reflects a higher, but still conservative allowable leak rate for MSIVs.

2. Section 3.6.1.4 and its associated Bases are amended to permit the deletion of the MSIVSS from the Technical Specifications. The revised analysis using Regulatory Guide 1.183 does not credit the operation of the MSIVSS.

3. Tables 3.3.2-1 and 3.6.3-1 be amended to permit the deletion of the MSIVSS valves and associated main steam line drain valves from the Technical Specifications. These lines will be cut and capped as a part of modifications associated with these proposed changes.

4. The Bases for Section 3.1.5 be amended to identify the use of the Standby Liquid Control System for controlling and maintaining long-term suppression pool water pH levels at 7 or above during the entire 30 day period of the postulated accident.

5. The Bases for Section 3.6.1.2 be amended to reflect that any MSIV exceeding the specified leakage limits will be repaired to less than or equal to 25 scfh.

6. The Bases for Section 3.7.2 be amended to reflect the change to TEDE and the use of 10CFR50.67.

7. The TS index is administratively amended to reflect the above changes.

In summary, these proposed Technical Specification changes are based on the results of revised offsite and control room operator dose calculations for the loss of coolant accident (LOCA), which is the most limiting Hope Creek Unit 1 Design Basis Accident (DBA), and using

an alternative source term in accordance with 10CFR50.67. In addition, control room inleakage has been increased to account for habitability requirements associated with NEI 99-03 (Draft). The dose calculations have been performed following the guidance contained in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

3.0 BACKGROUND

Each of the four main steam lines (MSLs) contains two quick-closing MSIVs, one located inside and one located outside the primary containment. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis loss-of-coolant accident (LOCA), or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur. Operating experience at various boiling water reactor (BWR) plants has indicated that degradation has occasionally occurred in the leak-tightness of MSIVs, and the specified low leakage limits are difficult to maintain.

The current Hope Creek Technical Specification allowable MSIV leak rate is limiting and routinely requires the repair and re-testing of the MSIVs. Furthermore, valve manufacturers have stated that leakage rates up to 200 scfh can occur without having a major valve defect. This unnecessary repair significantly impacts the maintenance workload, often contributes to outage extensions, and has in the past affected the operability of the MSIVs. Hope Creek outage planners routinely schedule several days of contingency to repair and re-test the MSIVs. In addition, the needless radiological exposure to maintenance personnel is inconsistent with the As Low As Reasonably Achievable (ALARA) requirements.

As a result of recurring problems with excessive leakage of MSIVs, the Nuclear Regulatory Commission (NRC) staff issued Regulatory Guide 1.96, which recommends the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified leakage limits. To meet this Regulatory Guide, Hope Creek installed a safety-related MSIVSS that is designed to eliminate the release of fission products through the closed MSIVs that would otherwise bypass the Filtration, Recirculation and Ventilation System (FRVS) after a LOCA. This is accomplished by pressurizing the sections of the MSLs between the inboard and the outboard MSIVs, and between the outboard MSIVs and the main steam stop valves, to a pressure above that of the reactor pressure vessel. Sealing gas is supplied from two independent primary containment instrument gas receivers. Leakage past the MSIVs is directed back into primary containment where it can be processed as a filtered release and reduce the potential contribution to offsite and control room doses.

The MSIVSS at Hope Creek (described in UFSAR Section 6.7) is a maintenance intensive system (several hundred man-hours per cycle are spent maintaining the MSIVSS and the MSIVs) that also affects the Primary Containment Instrument Gas (PCIG) system by allowing leaking steam/moisture into the dry gas system. As a result, Hope Creek proposes to delete the MSIVSS and to increase the Technical Specification allowable leakage rate for the MSIVs.

These changes are being considered now because of the significant advances that have been made in understanding the timing, magnitude and chemical form of fission product releases from a substantial core meltdown. Those advances are documented in the Alternate Source Term methodology of Regulatory Guide 1.183, and 10CFR50.67 and they result in lower BWR accident doses from releases. This is due primarily to the revised partitioning (aerosol, elemental, organic) of the iodine fission products and the resulting fraction of what comes out of

the reactor coolant and becomes airborne. Alternate Source Term methodology also involves the use of the TEDE criteria provided in 10CFR50.67, which sums dose contributions from inhalation and external exposure.

As noted above, knowledge of the more physically correct source timing and chemical form permits the use of more appropriate mitigation techniques. Specifically, natural forces such as gravitational settling of aerosol (particulates) have been credited inside the drywell and in portions of the main steam lines, which significantly reduces the amount of radionuclides that could escape from the containment and into the environment. Though, not credited for in our analysis, the Containment Spray system would operate post-LOCA for up to 24 hours. This would scrub released radionuclides from the containment atmosphere and into the suppression pool and would reduce the post-LOCA off-site and Control Room dose.

Once the containment sprays have been successful in sweeping the iodine to the suppression pool, the iodine must be retained in the water. To achieve this, the pH level of the suppression pool will be raised to 7 or above following the accident. Then due to injection of sodium pentaborate by the Standby Liquid Control System (SLCS), the pH will be maintained at 7 or above. The sodium pentaborate will be well mixed with the suppression pool water by the end of a 6-hour period as a result of reflooding the reactor vessel. This prevents significant fractions of the dissolved iodine from being converted to elemental iodine and then re-evolving to the containment atmosphere. During the course of the accident, the pH of the suppression pool can decrease due to radiolysis of reactor coolant and chloride-bearing electrical insulation, which would create acids. The pH of the suppression pool will be maintained at levels of 7 or above as a result of the SLCS introduction of sodium pentaborate, which produces a boron buffered environment. SLCS injection occurs in accordance with SAG-1. The current sodium pentaborate requirements as well as operability requirements in the TS remain unchanged.

The proposed elimination of the MSIVSS involves additional release paths that were not considered in the current LOCA analysis. Post-LOCA MSIV leakage activity can potentially be released to the environment through the south plant vent (SPV) or the turbine building louver (TBL). Atmospheric dispersion factors for these paths were calculated using ARCON96 computer code. In addition, all existing dispersion factors were revised using ARCON96 to provide a consistent basis for all release locations. The assumptions and applicable design inputs are documented in Calculation No. H-1-ZZ-MDC-1879 (Attachment 5).

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The Regulatory Requirements are contained in 10CFR50.67 and guidance has been provided in Regulatory Guide 1.183.

10CFR50.67 requires a licensee who seeks to revise its current accident source term in design basis radiological consequence analyses to apply for a license amendment under 10CFR50.90. The application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

The NRC will issue the amendment only if the applicant's analysis demonstrates with reasonable assurance that:

(i) An individual located at any point on the outer boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE for the duration of the accident.

Regulatory Guide 1.183 provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

5.0 TECHNICAL ANALYSIS

The MSIVSS is designed to eliminate the release of fission products through the MSIVs that would bypass FRVS filtration after a LOCA. This is accomplished by pressurizing the sections of the MSLs between the inboard and the outboard MSIVs, and between the outboard MSIVs and the main steam stop valves (MSSVs), to a pressure above that of the reactor pressure vessel (RPV). This would reduce the potential contribution to off-site and control room dose. Historically, this system has been susceptible to numerous failures and very costly repairs. In order to improve the performance of the power plant, both from the nuclear safety perspective and from the high cost of maintenance perspective, the MSIVSS will be deleted.

The elimination of the MSIVSS is proposed based on the implementation of AST and TEDE dose criteria. The characteristics of the AST (different in magnitude, timing, and chemical forms) and the revised dose calculation methodology became incompatible with many of the analysis assumptions and methods used in the Hope Creek Generating Station (HCGS) current licensing basis analyses. Therefore, the existing design input parameters and assumptions were assessed to determine their compatibility for the AST and integrated radiological response of the plant. Additionally, the design input parameters are validated to represent as-built design of the plant and performance of the safety related components credited in the analysis. The scope of the analysis is extended to evaluate the impact of proposed MSIV leakage increase from 46 scfh to 250 scfh and control room unfiltered inleakage from 10 cfm to 900 cfm.

The radiological dose analyses have been revised to document the evaluation of the Hope Creek design and configuration for its ability to limit the total calculated dose to less than the requirements of 10CFR50.67 without taking credit for the MSIVSS. The revised radiological analyses calculated the effects of the proposed allowable MSIV leak rate in terms of control room and off-site doses.

Calculations were performed to evaluate the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) doses due to:

- An increased value for CR unfiltered inleakage from 10 cfm to 900 cfm

- The deletion of the MSIVSS

- An increased value of MSIV leak rate from 46 scfh to 250 scfh

The doses were calculated using the Alternate Source Term (AST), Regulatory Guide (RG) 1.183 requirements, RADTRAD V3.02 computer code, and TEDE dose methodology. The following licensing basis post-LOCA release paths were analyzed:

1. Containment Leakage
2. Engineered Safety Feature (ESF) Leakage
3. Main Steam Isolation Valve (MSIV) Bypass Leakage

The RADTRAD V3.02 computer code was developed for the NRC for its use in control room habitability assessments. The RADTRAD code estimates transport and removal of radionuclides and doses at selected receptors. In addition, the code can account for a reduction in the quantity of radioactive material due to containment sprays, natural deposition, filters, and other natural engineered safety features. The EAB, LPZ, and CR doses were calculated using the release paths such as Containment Leakage, ESF Leakage, and MSIV leakage using the as-built design inputs/assumptions and the guidance in Regulatory Guide 1.183. The structures, systems, and components capable of performing their safety functions during and following a safe shutdown earthquake (SSE) are credited in the analysis.

The RADTRAD V3.02 code is benchmarked using the HABIT1.0 code, which is currently utilized for HCGS design basis accident (DBA) analyses. The current HABIT1.0 TID-14844 release model was re-run using the RADTRAD V3.02 code with the consistent source terms, transport mechanisms, and dose conversion factors to determine the ability of RADTRAD code to produce consistent results with accuracy of $\pm$ 2%.

The above release paths were analyzed with the following assumptions to demonstrate additional conservatisms in the AST analysis:

1. The safety related drywell and torus sprays are not credited in this analysis for removal of elemental and aerosol airborne activity.
2. The Control Room Emergency Filtration (CREF) system is assumed to start at 30 minutes after a LOCA.
3. The FRVS vent exhaust flow rate is increased by 10% and FRVS & CREF recirculation flow rates are decreased by 10% to maximize the doses at various receptor locations.

Additional assumptions and design inputs are provided in Calculation No. H-1-ZZ-MDC-1880 (Attachment 4). Table 1 of this attachment shows the calculated dose exposures from these analyses and the allowable TEDE limits contained in 10CFR50.67 and Regulatory Guide 1.183 Table 6.

TABLE 1

DOSE ANALYSIS

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (Rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Containment Leakage	4.50E-01	3.41E-01(3.2 hr)	1.10E-01
ESF Leakage	2.85E-01	3.51E-02 (0 hr)	1.19E-02
MSIV Leakage	3.48E+00	1.92E+00 (9.3 hr)	3.67E-01
Containment Purge	0.00E+00	0.00E+00	0.00E+00
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	0.00E+00	0.00E+00	0.00E+00
CR Filter Shine	2.46E-03*	0.00E+00	0.00E+00
Total	**4.22E+00**	**2.21E+00**	**4.89E-01**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**

*CR filter shine dose due to the CR unfiltered inleakage of 1000 cfm with RADTRAD default nuclide inventory file (NIF) will bound that due to the CR unfiltered inleakage of 900 cfm with the plant-specific NIF.

The calculated values in the revised analysis demonstrate that an MSIV leakage rate of 150 scfh per main steam line (not to exceed 250 scfh total for all four main steam lines), an increase in control room unfiltered inleakage to a value of 900 cfm and with no credit taken for the MSIVSS results in an acceptable increase in the dose exposures for the control room, Exclusion Area Boundary (EAB), and Low Population Zone (LPZ). The revised post-LOCA doses remain within the allowable limits.

6.0 REGULATORY ANALYSIS

The revised dose analysis was performed using AST and TEDE dose criteria in accordance with the guidance provided in Regulatory Guide 1.183. The RADTRAD V3.02 computer code was utilized to perform the calculations. All the assumptions and applicable design inputs are documented in Calculation No. H-1-ZZ-MDC-1880.

The conclusion of the analysis is that with the proposed increase in MSIV leakage rate, deletion of MSIVSS, increase in control room unfiltered inleakage, and using AST and TEDE dose criteria as the basis of the analysis, the HCGS will be in compliance with the requirements of 10CFR50.67.

Also, in accordance with Regulatory Guide 1.183, Section 6, (Assumptions For Evaluating The Radiation Doses For Equipment Qualification), the NRC staff is assessing the effect of increased cesium releases on EQ doses to determine whether licensee action is warranted. Until such time as this generic issue is resolved, licensees may use either the AST or TID-14844 assumptions for performing the required EQ analyses. However, no plant modifications are required to address the impact of the difference in source term characteristics (i.e., AST vs. TID-14844) on EQ doses pending the outcome of the evaluation of the generic issue.

Therefore, no further evaluation, other than reviewing current EQ status, has been taken. Equipment important to safety remains unaffected by the implementation of AST.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

PSEG Nuclear LLC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment 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 proposed changes to TS Sections 3.6.1.2 and 3.6.1.4 do not involve a change to structures, components, or systems that would affect the probability of an accident previously evaluated in the Hope Creek Updated Final Safety Analysis Report (UFSAR).

   The proposed changes involve eliminating the MSIVSS requirements from the TS. As described in Section 6.7 of the UFSAR, the MSIVSS is manually initiated after a design basis Loss of Coolant Accident (LOCA). Since the MSIVSS is operating only after an accident has occurred, these proposed changes have no effect on the probability of an accident. MSIV leakage and operation of the MSIVSS do not affect the precursors of any design basis accidents. Analysis of the affects of the proposed changes do, however, result in acceptable radiological consequences for the design basis LOCA previously evaluated in Section 15.6.5 of the UFSAR. The revised analysis calculated offsite and control room operator dose consequences using an alternative source term and following the guidance provided in Regulatory Guide 1.183. Also, no credit was taken for the MSIVSS in this revised analysis. The results demonstrate that dose consequences remain within the allowable limits.

   A plant-specific radiological analysis has been performed in accordance with Regulatory Guide 1.183 to assess the effects of the proposed increase to the allowable MSIV leakage rate and CR inleakage, in terms of CR and off-site doses following a postulated design basis LOCA. The radiological analysis uses an alternative source term and is performed in accordance with 10CFR50.67.

   The analysis results demonstrate that dose contributions from the proposed MSIV leakage rate limit of 150 scfh per steam line, not to exceed a total of 250 scfh for all four main steam lines, from the proposed deletion of the MSIVSS, and increase to 900 cfm of control room inleakage, result in values for the off-site doses and MCR doses that are within the acceptance criteria specified in 10CFR50.67

   Table 1, provided in Section 5.0 of this attachment, contains the post-LOCA dose results that were based on Calculation No. H-1-ZZ-MDC-1880. It can be concluded from these results that with the changes proposed by this amendment application, the calculated dose increase for the control room, Exclusion Area

Boundary and Low Population Zone following a LOCA remain below the allowable regulatory limits.

The proposed change to TS Tables 3.3.2-1 and 3.6.3-1 involves the deletion of MSIVSS valves and associated main steam line drain valves from the list of primary containment isolation valves. This proposed change is consistent with the proposed deletion of the MSIVSS. The MSIVSS lines and main steam line drain valves that are connected to the main steam piping will be capped and welded closed to assure primary containment integrity is maintained. The welding and post weld examination procedures will be in accordance with American Society of Mechanical Engineers (ASME) Code, Section III requirements. These welded caps will be periodically tested as part of the Containment Integrated Leak Rate Test (CILRT). This proposed change does not involve an increase in the probability of equipment malfunction previously evaluated in the UFSAR. This proposed change has no effect on the consequences of an accident since the MSIVSS lines and associated main steam line drain valves will be capped and welded closed, thus assuring that the containment integrity, isolation, and leak test capability are not compromised.

Therefore, as discussed above, the proposed changes do not involve a significant increase in the probability or consequences from any 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.

The proposed change to increase the allowed MSIV leakage rate does not affect the operability of the MSIVs and will not inhibit the capability of the MSIVs to effectively isolate the primary containment and therefore does not create any new or different kind of accident from any accident previously evaluated. The proposed change to eliminate the MSIVSS does not create the possibility of a new or different kind of accident from any accident previously evaluated because the removal of the MSIVSS does not affect any of the remaining Hope Creek systems, and the LOCA has been re-analyzed using an alternative source term and the guidance provided in Regulatory Guide 1.183. The associated proposed change to delete the MSIV Sealing Isolation Valves and associated Main Steam Line Drain Valves from TS Tables 3.3.2-1 and 3.6.3-1 does not create the possibility of a new or different kind of accident, since the affected main steam piping will be welded and/or capped closed to assure that the primary containment integrity, isolation, and leak testing capability are not compromised. Also, the increase in CR inleakage to 900 cfm does not create the possibility of a new or different kind of accident, since this does not affect any plant structures, systems or components.

Therefore, as discussed above, the proposed changes do not create the possibility for any new or different kind of accident from any accident previously evaluated.

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

Response: No.

The proposed change to TS Section 3.6.1.2 to increase the MSIV allowable leakage does not involve a significant reduction in the margin of safety. As discussed in the current Bases for TS Section 3/4.6.1.2, the allowable leak rate limit specified for the MSIVs is used to quantify a maximum amount of leakage assumed to bypass primary containment in the LOCA radiological analysis. Accordingly, results of the re-analysis using Regulatory Guide 1.183 supporting these proposed changes are evaluated against the dose limits contained in 10CFR50.67. As discussed above, sufficient margin relative to the regulatory limits is maintained as a result of this revised analysis (see Table 1 in Section 5.0).

The proposed change to eliminate the MSIVSS from TS does not reduce the margin of safety. In fact, the revised analysis uses an alternative source term and the guidance provided in Regulatory Guide 1.183 but does not take credit for the operation of the MSIVSS. The results of the offsite and control room operator dose calculations demonstrate that the consequences of the design basis-limiting event are within the acceptance criteria specified in 10CFR50.67 and Regulatory Guide 1.183.

In addition, the revised calculation shows that the radiological consequences of the proposed MSIV leakage rate limit of 150 scfh per main steam line, not to exceed a total of 250 scfh for all four main steam lines and increase to 900 cfm in CR inleakage, would not exceed the regulatory limits.

The proposed change to delete the MSL Drain and MSIVSS Isolation Valves from TS Tables 3.3.2-1 and 3.6-3-1 does not reduce the margin of safety. Welded capped closure of the MSIV Sealing lines assures that the primary containment integrity and leak testing capability are not compromised. These welded caps will be periodically leak tested as part of the CILRT. Therefore, the proposed deletion of the MSIVSS and MSL Drain Isolation Valves does not involve a reduction in a margin of safety.

Accordingly, based on the above reasons, the proposed changes do not involve a significant reduction in a margin of safety.

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

8.0 ENVIRONMENTAL CONSIDERATION

PSEG Nuclear LLC has reviewed the proposed TS changes against the criteria of 10CFR51.22 for environmental considerations. 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 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENT

This submittal requests an amendment to the TS, to permit an increase in the allowable leak rate for the MSIVs and to delete the MSIVSS. The NRC has approved similar changes in Amendment 103 for First Energy Nuclear Operating Company's Perry Nuclear Power Plant. Also, in Amendment 145 for Entergy Operations, Inc.'s Grand Gulf Nuclear Station, Unit 1 the NRC approved an increase in the allowable MSIV leakage.

10.0 REFERENCES

1. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors", July 2000.
2. S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation", NUREG/CR-6604, USNRC, April 1998.
3. 10CFR50.67, "Accident Source Term".
4. HCCALC H-1-ZZ-MDC-1880, Rev 0, Post-LOCA EAB, LPZ, and CR Doses – Alternate Source Term Analysis.
5. HCCALC H-1-ZZ-MDC-1879, Rev 0, Control Room Chi/Qs For FRVS Vent, RB Truck Bay, TB Louvers, and SPV.
6. HCGS Technical Specifications.
7. NEI 99-03 (Draft), February 2001, Control Room Habitability Assessment Guidance.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF- 57
DOCKET NO. 50-354
REVISIONS TO THE TECHNICAL SPECIFICATIONS (TS)

TECHNICAL SPECIFICATION PAGES WITH PROPOSED CHANGES

The following Technical Specifications for Facility Operating License No. NPF-57 are affected by this change request:

Technical Specification	Page
Index	xi & xix
Table 3.3.2-1	3/4 3-17 & 3-18
3.6.1.2	3/4 6-2 & 6-3
3.6.1.4	3/4 6-7
Table 3.6.3-1	3/4 6-19, 6-20 & 6-42
Bases 3/4.1.5	B 3/4 1-5
Bases 3/4.6.1.2 & 6.1.4	B 3/4 6-1 & 6-2
Bases 3/4.7.2	B 3/4 7-1

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

This table notation identifies which valves, in an actuation group, are closed by a particular trip signal. If all valves in the group are closed by the trip signal, only the valve group number will be listed. If only certain valves in the group are closed by the trip signal, the valve group number will be listed followed by, in parentheses, a listing of which valves are closed by the trip signal.

delete

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - 1) Low Low, Level 2	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 2, 8, 9, 12, 13, 14, 15 (HV-5154, HV-5155), 17, 18
2) Low Low Low, Level 1	10, 11, 15(HV-5126 A&B, HV-5152 A&B, HV-5147, HV-5148 HV-5162), 16
b. Drywell Pressure - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18
c. Reactor Building Exhaust Radiation - High	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 12, 13, 14, 15, 17 (HV-5161), 18
d. Manual Initiation	1 (HV-5834A, HV-5835A, HV-5836A, HV-5837A), 8, 9, 10, 11, 12, 13, 14, 15, 16, 17 (HV-5161), 18
2. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	19
b. Drywell Pressure - High	19
c. Refueling Floor Exhaust Radiation - High	19
d. Reactor Building Exhaust Radiation - High	19
e. Manual Initiation	19

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

delete

TRIP FUNCTION	VALVES CLOSED BY SIGNAL
3. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	1 (HV-F022A, B, C & D, HV-F028A, B, C & D, ~~HV-F067A, B, C & D,~~ HV-F016, HV-F019)
b. Main Steam Line Radiation - High, High	2
c. Main Steam Line Presure - Low	1 (as above)
d. Main Steam Line Flow - High	1 (as above)
e. Condenser Vacuum - Low	1 (as above)
f. Main Steam Line Tunnel Temperature - High	1 (as above)
g. Manual Initiation	1 (as above), 2, 17 (SV-J004A-1, 2, 3, 4 & 5)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. RWCU Δ Flow - High	7
b. RWCU Δ Flow - High, Timer	7
c. RWCU Area Temperature - High	7

150 scfh per main steam line and less than or equal to 250

CONTAINMENT SYSTEMS

PRIMARY CONTAINMENT LEAKAGE

leakage rate corrected to 1 Pa, 48.1 psig

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate (Type A test) in accordance with the Primary Containment Leakage Rate Testing Program.

b. A combined leakage rate in accordance with the Primary Containment Leakage Rate Testing Program for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests.

c. *Less than or equal to ~~46.0~~ scfh combined through all four main steam lines when tested at 5 psig (~~seal system Δp~~).

d. A combined leakage rate of less than or equal to 10 gpm for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1, when tested at 1.10 Pa, 52.9 psig.

e. A combined leakage rate of less than or equal to 10 gpm for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment, when tested at 1.10 Pa, 52.9 psig Δp.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:
With:

a. The measured overall integrated primary containment leakage rate (Type A test) not in accordance with the Primary Containment Leakage Rate Testing Program, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests not in accordance with the Primary Containment Leakage Rate Testing Program, or

c. The measured leakage rate exceeding ~~46.0~~ scfh combined through all four main steam lines, or

150 scfh per main steam line or exceeding 250

*Exemption to Appendix "J" of 10 CFR 50.

150 scfh per main steam line and less than or equal to 250

ACTION (Continued)

d. The measured combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 exceeding 10 gpm, or

e. The measured combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment exceeding 10 gpm,

restore:

a. The overall integrated leakage rate(s) (Type A test) to be in accordance with the Primary Containment Leakage Rate Testing Program, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves*, valves which form the boundary for the long-term seal of the feedwater lines, and other valves which are hydrostatically tested per Table 3.6.3-1, subject to Type B and C tests to be in accordance with the Primary Containment Leakage Rate Testing Program, and

c. The leakage rate to less than or equal to ~~46.0~~ scfh combined through all four main steam lines, and

d. The combined leakage rate for all containment isolation valves which form the boundary for the long-term seal of the feedwater lines in Table 3.6.3-1 to less than or equal to 10 gpm, and

e. The combined leakage rate for all other penetrations and containment isolation valves in hydrostatically tested lines in Table 3.6.3-1 which penetrate the primary containment to less than or equal to 10 gpm,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2.a The primary containment leakage rates shall be demonstrated in accordance with the Primary Containment Leakage Rate Testing Program for the following:

1. Type A test.

2. Type B and C tests (including air locks).

b. DELETED.

c. DELETED.

* Exemption to Appendix "J" of 10 CFR 50.

delete

CONTAINMENT SYSTEMS

MSIV SEALING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV sealing system (MSIVSS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With one MSIV sealing system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 Each MSIV sealing system subsystem shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each testable motor-operated valve except the Main Steam Stop Valves (MSSVs) through at least one complete cycle of full travel.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor-operated valve including the Main Steam Stop Valves (MSSVs) not testable during operation through at least one complete cycle of full travel.

c. At least once per 18 months by performance of a functional test of the subsystem throughout its operating sequence, and verifying that each interlock and timer operates as designed and each automatic valve actuates to its correct position.

d. By verifying the control instrumentation to be OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,
2. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
3. CHANNEL CALIBRATION at least once per 18 months.

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TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
A. Automatic Isolation Valves				
1. Group 1 - Main Steam system				
(a) Main Steam Isolation Valves (MSIVs)				
Inside:				M-41-1
Line A HV-F022A (AB-V028)	P1A	5	1	
Line B HV-F022B (AB-V029)	P1B	5	1	
Line C HV-F022C (AB-V030)	P1C	5	1	
Line D HV-F022D (AB-V031)	P1D	5	1	
Outside:				
Line A HV-F028A (AB-V032)	P1A	5	1	
Line B HV-F028B (AB-V033)	P1B	5	1	
Line C HV-F028C (AB-V034)	P1C	5	1	
Line D HV-F028D (AB-V035)	P1D	5	1	
(b) Main Steam Line Drain Isolation				M-41-1
Inside: HV-F016 (AB-V039)	P12	30	3	
Outside:				
Line A HV-F067A (AB-V059)	P1A	45	1	
Line B HV-F067B (AB-V060)	P1B	45	1	
Line C HV-F067C (AB-V061)	P1C	45	1	
Line D HV-F067D (AB-V062)	P1D	45	1	
HV-F019 (AB-V040)	P12	30	3	

delete

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	PENETRATION NUMBER	MAXIMUM ISOLATION TIME (Seconds)	NOTE(S)	P&ID
~~(c) MSIV Sealing System Isolation Valves~~				~~M-72-1~~
~~Outside:~~				
~~Line A HV-5834A (KP-V010)~~	~~P1A~~	~~45~~	~~1~~	
~~Line B HV-5835A (KP-V009)~~	~~P1B~~	~~45~~	~~1~~	
~~Line C HV-5836A (KP-V008)~~	~~P1C~~	~~45~~	~~1~~	
~~Line D HV-5837A (KP-V007)~~	~~P1D~~	~~45~~	~~1~~	
		delete		
2. Group 2 - Reactor Recirculation Water Sample System				
(a) Reactor Recirculation Water Sample Line Isolation Valves				M-43-1
Inside: BB-SV-4310	P17	15	3	
Outside: BB-SV-4311	P17	15	3	
3. Group 3 - Residual Heat Removal (RHR) System				
(a) RHR Suppression Pool Cooling Water & System Test Isolation Valves				M-51-1
Outside:				
Loop A: HV-F024A (BC-V124)	P212B	180	11	
HV-F010A (BC-V125)	P212B	180	11	
Outside:				
Loop B: HV-F024B (BC-V028)	P212A	180	11	
HV-F010B (BC-V027)	P212A	180	11	
(b) RHR to Suppression Chamber Spray Header Isolation Valves				M-51-1
Outside:				
Loop A: HV-F027A (BC-V112)	P214B	75	3	
Loop B: HV-F027B (BC-V015)	P214A	75	3	

TABLE 3.6.3-1

PRIMARY CONTAINMENT ISOLATION VALVES

NOTES

NOTATION

delete

1. Main Steam Isolation Valves are sealed with a seal system that maintains a positive pressure of 8 psig above reactor pressure. l Leakage is inleakage and is not added to 0.60 La allowable leakage.*

2. Containment Isolation Valves are sealed with a water seal from the HPCI and/or RCIC system to form the long-term seal boundary of the feedwater lines. The valves are tested with water at 1.10 Pa, 52.9 psig, to ensure the seal boundary will prevent by-pass leakage. Seal boundary liquid leakage will be limited to 10 gpm.

3. Containment Isolation Valve, Type C gas test at Pa, 48.1 psig. Leakage added to 0.60La allowable leakage.

4. Containment Isolation Valve, Type C water test at 1.10 Pa, 52.9 psig delta P. Leakage added to 10 gpm allowable leakage.

5. Containment boundary is discharge nozzle of relief valve, leakage tested during Type A test.*

6. Drywell and suppression chamber pressure and level instrument root valves and excess flow check valves, leakage tested during Type A.*

7. Explosive shear valves (SE-V021 through SE-V025) not Type C tested.*

8. Surveillances to be performed per Specification 3.6.1.8.

9. All valve I.D. numbers are preceded by a numeral 1 which represents an Unit 1 valve.

10. The reactor vessel head seal leak detection line (penetration J5C) excess flow check valve (BB-XV-3649) is not subject to OPERABILITY testing. This valve will not be exposed to primary system pressure except under the unlikely conditions of a seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source; therefore, this valve need not be OPERABILITY tested.

11. Containment Isolation Valve(s) are not Type C tested. Containment by-pass leakage is prevented since the line terminates below the minimum water level in the suppression chamber and the system is a closed system outside Primary Containment. Refer to Specification 4.0.5.

*Exemption to Appendix J of 10 CFR Part 50.

rate, solution concentration or boron equivalent to meet the ATWS Rule must not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

"Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron control equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment)."

The described minimum system parameters (82.4 gpm, 13.6 percent concentration and natural boron equivalent) will ensure an equivalent injection capability that exceeds the ATWS Rule requirement. The stated minimum allowable pumping rate of 82.4 gallons per minute is met through the simultaneous operation of both pumps.

The standby liquid control system will also provide the capability to raise and maintain the long-term post-accident coolant inventory pH levels to 7 or above. This will prevent significant fractions of the dissolved iodine from being converted to elemental iodine and then re-evolving to the containment atmosphere.

Add paragraph

1. CENPD-284-P-A, "Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification," July, 1996.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak ... This restriction, in conjun... site boundary radiat... accident conditior...

If the leakage rate on a main steam line exceeds the requirements of Technical Specification 3.6.1.2.c. (150 scfh), the leakage rate for that line will be restored to less than or equal to 25 scfh (when tested at 5 psig and corrected to Pa) prior to plant restart.

In high rad... controlled in a ... means to verify ... Surveillance Requ... restricted in accor... and/or plant procedur... is performed before restarting from each ... Therefore, the probability of misalignment of these components, once they have been verified to be in the proper position, is low.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the design basis LOCA maximum peak containment accident pressure of 48.1 psig, P_a. As an added conservatism, the measured overall integrated leakage rate (Type A test) is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the Primary containment Leakage Rate Testing Program.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the Primary Containment Leakage Rate Testing Program. Only one closed door in each air lock is required to maintain the integrity of the containment.

Deleted

~~3/4.6.1.4 MSIV SEALING SYSTEM~~

~~Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser~~

delete

===

~~3/4.6.1.4 MSIV SEALING SYSTEM (Continued)~~

~~remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The sealing system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.~~

delete

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 48.1 psig in the event of a LOCA. A visual inspection in accordance with the Primary Containment Leakage Rate Testing Program is sufficient.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of 48.1 psig does not exceed the design pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 3 psid. The limit of -0.5 to +1.5 psig for initial positive containment pressure will limit the total pressure to 48.1 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis. The 135°F average temperature is conducive to normal and long term operation.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 500 hours/365 days limit for the operation of the purge valves and the 6" nitrogen supply valve during plant Operational Conditions 1, 2 and 3 is intended to reduce the probability of a LOCA occurrence during the above operational conditions when the applicable combination of the above valves are open.

Blow-out panels are installed in the CPCS ductwork to provide additional assurance that the FRVs will be capable of performing its safety function subsequent to a LOCA.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the station service water and the safety auxiliaries cooling systems ensures that sufficient cooling capacity is available for continued operation of the SACS and its associated safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters and humidity control instruments OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less ~~whole body, or its equivalent~~ total effective dose equivalent (TEDE). This limitation is consistent with the requirements of ~~General Design Criteria 19 of Appendix "A"~~ 10 CFR Part ~~50~~ 50.67, "Accident Source Term."

3/4.7.3 FLOOD PROTECTION

The requirement for flood protection ensures that facility flood protection features are in place in the event of flood conditions. The limit of elevation 10.5' Mean Sea Level is based on the elevation at which facility flood protection features provide protection to safety related equipment.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

CALCULATION NO: H-1-ZZ-MDC-1880

Post-LOCA EAB, LPZ, and CR Doses – Alternate Source Term Analysis.

CALC NO.: H-1-ZZ-MDC-1880 REVISION: 0IR1	CALCULATION COVER SHEET	Page 1 of 68
CALC. TITLE:	Post-LOCA EAB, LPZ, and CR Doses – Alternate Source Term Analysis	

# SHTS (CALC):	68	# ATT / # SHTS:	1	# IDV/50.59 SHTS:	1/0	# TOTAL SHTS:	70

CHECK ONE:

☐ FINAL ☒ INTERIM (Proposed Plant Change) ☐ FINAL (Future Confirmation Req'd) ☐ VOID

SALEM OR HOPE CREEK: ☐ Q – LIST ☒ IMPORTANT TO SAFETY ☐ NON-SAFETY RELATED

HOPE CREEK ONLY: ☒Q ☐Qs ☐Qsh ☐F ☐R

☒ STATION PROCEDURES IMPACTED, IF SO CONTACT SYSTEM MANAGER

☐ CDs INCORPORATED (IF ANY): ____________

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

N/A

PURPOSE:

The purpose of this calculation is to determine the EAB, LPZ, and control room doses for Hope Creek Generating Station (HCGS) due to the increased CR unfiltered inleakage from 10 cfm to 900 cfm, the deletion of MSIV Sealing System (MSIVSS), and the increased MSIV leakage from 46 scfh to 250 scfh. The analysis is performed using the Alternate Source Term (AST), the guidance in the Regulatory Guide 1.183, and the TEDE dose criteria. The V&V of RADTRAD3.02 computer code is performed using the HABIT1.0 code, which is currently used for the licensing basis analyses at the Hope Creek and Salem plants.

The 10CFR50.59 evaluation for DCP 4EC-3513, Package No. 1 applies to this documentation which is CD P606.

CONCLUSIONS:

The results of analyses in Section 8 indicate that the main steam sealing system can be safely eliminated along with the increased MSIV leakage of 250 scfh and control room unfiltered inleakage of 900 cfm using the AST and guidance in the Regulatory guide 1.183. Adherence to guidance in the RG 1.183 and use of the specific values and limits contained in the technical specifications and as-built post-accident performance of safety grade ESF functions provide the assurance for sufficient safety margin, including a margin to account for analysis uncertainties in the proposed uses of an AST and the associated facility modifications and changes to procedures. The V&V of RADTRAD3.02 code demonstrates that RADTRAD produces the identical results within ± 2% margin of error compared to the HABIT1.0 results.

Confirmation Required: The following Design Inputs are to be confirmed:

Design Input 5.3.1.3, "Isotopic Core Inventory," page 13 & Table 16 (Order No. 80028003).
Design Input 5.4.5, "Suppression Pool Water pH," page 17 (Order No. 80028003).

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	Gopal J. Patel/NUCORE	05/15/01
PEER REVIEWER/COMPANY NAME:	John Duffy/PSEG	05/16/01
VERIFIER/COMPANY NAME:	John Duffy/PSEG	05/16/01
PSEG SUPERVISOR APPROVAL:	Robert DeNight/PSEG	05/16/01

REVISION HISTORY

Revision	Issue Date	Revision Description
0IR0	5/7/01	Initial Issue.
0IR1	5/16/01	Revised due to incorporation of the preliminary plant-specific core inventory, which will be confirmed via Order No. 80028003. The CR inleakage value was reduced to 900 cfm from 1000 cfm to offset the impact of preliminary core inventory on the CR dose.

PAGE REVISION INDEX

PAGE	REV	PAGE	REV	PAGE	REV
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3	1	44	0	Attachment C	1
4	1	45	0	Attachment D	1
5	1	46	0	Attachment E	1
6	0	47	0	Attachment F	1
7	0	48	0	Attachment G	1
8	0	49	0	Attachment H	0
9	0	50	0	Attachment I	0
10	0	51	0	Attachment J	0
11	0	52	0	Attachment K	0
12	0	53	0	Attachment L	0
13	1	54	0	Attachment M	0
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16	0	57	0		
17	0	58	0		
18	0	59	0		
19	0	60	1		
20	1	61	0		
21	0	62	1		
22	0	63	1		
23	0	64	0		
24	0	65	1		
25	1	66	1		
26	0	67	0		
27	0	68	1		
28	0				
29	0				
30	1				
31	0				
32	0				
33	0				
34	0				
35	0				
36	0				
37	0				
38	0				
39	0				
40	0				
41	0				

TABLE OF CONTENTS

Diskettes with the following electronic files
Calculation No: H-1-ZZ-MDC-1880

1.0 PURPOSE:

The purpose of this calculation is to evaluate the Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and Control Room (CR) Post-LOCA doses for Hope Creek Generating Station due to:

- An assumed increase of CR unfiltered inleakage from 10 cfm to 900 cfm,
- The deletion of Main Steam Isolation Valve (MSIV) Sealing System (MSIVSS), and
- An allowable increase of MSIV leakage from 46 scfh to 250 scfh.

The final results of the analyses are shown in Section 8.0 of this calculation. The doses are calculated using the Alternate Source Term (AST), Regulatory Guide (RG) 1.183 requirements, NRC sponsored RADTRAD3.02 computer code, and Total Effective Dose Equivalent (TEDE) dose methodology. Additionally, the RADTRAD3.02 code is benchmarked using the HABIT1.0 code using the TID release models with the consistent source terms, transport mechanisms, and dose conversion factors to demonstrate the ability of RADTRAD code to produce consistent results with an accuracy of ±2%. The comparison of results is shown in Section 8 and computer runs are shown in Attachment O.

2.0 SCOPE:

The scope of this evaluation covers the anticipated dose consequences of a Post-LOCA scenario for the HCGS. This calculation is being performed in support of Design Change Package (DCP) 4EC-3513, MSIV Steam Sealing System Deletion. As part of this analysis, the following licensing basis post-LOCA release paths are analyzed:

1. Containment Leakage.
2. Engineered Safety Feature (ESF) Leakage.
3. Main Steam Isolation Valve (MSIV) Bypass Leakage.

3.0 ANALYTICAL APPROACH

The elimination of the MSIV sealing system (MSIVSS) is proposed based on the implementation of AST and TEDE dose criteria. The characteristics of the AST (different in magnitude, timing, and chemical forms) and the revised dose calculation methodology became incompatible with many of the analysis assumptions and methods currently used in the current licensing basis analyses for HCGS. Therefore, the existing design input parameters and assumptions were assessed to determine their compatibility for the AST and integrated radiological response of the plant. Additionally, the design input parameters are validated to represent as-built design of the plant and performance of the safety grade components credited in the analysis.

The RADTRAD3.02 computer code (Ref. 10.2) was developed for the U.S. Nuclear Regulatory Commission Office Of Nuclear Reactor Regulation for use in control room habitability assessments. The

RADTRAD code estimates transport and removal of radionuclides and doses at selected receptors. In addition, the code can account for a reduction in the quantity of radioactive material due to containment sprays, natural deposition, filters, and other natural engineered safety features. The EAB, LPZ, and CR doses are calculated using the release paths such as containment leakage, ESF leakage, and MSIV leakage using the as-built design inputs/assumptions and guidance in the Regulatory Guide 1.183 (Ref. 10.1). The structure, system, and components capable of performing their safety functions during and following a safe shutdown earthquake (SSE) are credited in the analysis.

4.0 ASSUMPTIONS

The following assumptions used in evaluating the offsite and control room doses resulting from a Loss of Coolant Accident (LOCA) are based on the requirements in the Regulatory Guide 1.183 (Ref. 10.1). These assumptions become the design inputs in Sections 5.3 through 5.7 and are incorporated in the analyses.

4.1 Source Term Assumptions

Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Positions (RGP) 3.1 through 3.4 of Reference 10.1 as follows:

4.2 Core Inventory

The assumed inventory of fission products in the reactor core and available for release to the containment is based on the maximum power level of 3,458 MWt corresponding to current fuel enrichment and fuel burnup, which is 1.05 times the current licensed rated thermal power of 3,293 MWt for HCGS (Reference 10.6.9). The assumed core inventory is shown in Table 1 of Design Input 5.3.1.3.

4.3 Release Fractions and Timing

The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage for a Design Basis Accident (DBA) LOCA are listed in Table 3 of Design Input 5.3.1.5. These fractions are applied to the equilibrium core inventory described in Design Input 5.3.1.3 (Ref. 10.1, Tables 1 & 4).

4.4 Radionuclide Composition

The elements in each radionuclide group to be considered in design basis analyses are shown in Table 2 of Design Input 5.3.1.4 (Ref. 10.1, RGP 3.4).

4.5 Chemical Form

A pH value of 7.0 or greater for the suppression pool water inventory is assumed. Consequently, the chemical forms of radioiodine released to the containment can be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide (Ref. 10.1, RGP 3.5 and A.2). These are shown in Design Inputs 5.3.1.7. With the exception of elemental and organic iodine and noble gases, fission products are assumed to be in particulate form (Ref. 10.1, RGP 3.5 and A.2).

4.6 Assumptions on Activity Transport in Primary Containment

4.6.1 The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment.

4.6.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment is credited using the RADTRAD3.02 Powers model for aerosol removal coefficient with a 10-percentile probability (Ref. 10.1 RGP A.3.2 & 10.2).

4.6.3 The primary containment is assumed to leak at the allowable Technical Specification peak pressure leak rate for the first 24 hours (Ref. 10.1, RGP A.3.7). For HCGS, this leakage is reduced to 50% of its TS value after the first 24 hours based on the post-LOCA containment pressure (Ref. 10.15) as shown in design input 5.3.2.5.

4.6.4 The HCGS drywell and suppression chamber may be purged for up to 500 hrs per year (Ref. 10.6.18). Normally, the containment is purged at <25% power level before or during a drywell entry in an outage. Per RG 1.183, RGP A.7, the radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. However, HCGS has a safety grade hydrogen recombination system to control the post-accident combustible gas (Ref. 10.41 & 10.42). The post-LOCA containment pressure is reduced to less than 31 psia within a few days (Ref 10.15). Containment purging is not required for the combustible gas or pressure control measure within 30 days of the LOCA. Therefore, the release from containment purging is not analyzed.

4.7 Offsite Dose Consequences

The following assumptions are used in determining the TEDE for a maximum exposed individual at EAB and LPZ locations:

4.7.1 The offsite dose is determined in the TEDE, which is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure from all radionuclides that are significant with regard to dose consequences and the released radioactivity (Ref. 10.1, RGP 4.1.1, Ref 10.7). The RADTRAD3.02 computer code (Ref. 10.2) performs this summation to calculate the TEDE.

4.7.2 The offsite dose analysis is performed using the RADTRAD3.02 code (Ref. 10.2), which uses the Committed Effective Dose (CED) Conversion Factors for inhalation. (Ref. 10 .1, RGP 4.1.2, Refs. 10.7 & 10.8).

4.7.3 Since RADTRAD3.02 calculates Deep Dose Equivalent (DDE) using whole body submergence in semi-infinite cloud with appropriate credit for attenuation by body tissue, the DDE can be

assumed nominally equivalent to the effective dose equivalent (EDE) from external exposure. Therefore, the code uses DDE in lieu of EDE in determining TEDE (Ref. 10.1, RGP 4.1.4, and Ref 10.8).

4.7.4 The maximum EAB TEDE for any two-hour period following the start of the radioactivity release is determined and used in determining compliance with the dose acceptance criteria in 10 CFR 50.67 (Ref. 10.1, RGP 4.1.5 & RGP 4.4, and Ref. 10.4).

EAB Dose Acceptance Criteria: 25 Rem TEDE (50.67(b)(2)(i))

4.7.5 TEDE is determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and is used in determining compliance with the dose criteria in 10 CFR 50.67 (Refs. 10.1, RGP 4.1.6 and RGP 4.4 & Ref. 10.4).

LPZ Dose Acceptance Criteria: 25 Rem TEDE (50.67(b)(2)(ii))

4.7.6 No correction is made for depletion of the effluent plume by deposition on the ground (Ref. 10.1, RGP 4.1.7).

4.7.7 The breathing rates used for persons at offsite locations is given in Reference 10.1, RGPs 4.1.3 & 4.4. These rates are incorporated in design input 5.7.3.

4.8 Control Room Dose Consequences

The following guidance is used in determining the TEDE for maximum exposed individuals located in the control room:

4.8.1 The CR TEDE analysis considers the following sources of radiation that will cause exposure to control room personnel (Ref. 10.1, RGP 4.2.1). See applicable Design Inputs 5.6.1 through 5.6.13.

- Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the post-accident radioactive plume released from the facility (via CR air intake),
- Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope (via CR unfiltered inleakage),
- Radiation shine from the external radioactive plume released from the facility (external airborne cloud),
- Radiation shine from radioactive material in the reactor containment (containment shine dose), and

- Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters (CR filter shine dose).

4.8.2 The radioactivity releases and radiation levels used for the control room dose is determined using the same source term, transport, and release assumptions used for determining the exclusion area boundary (EAB) and the low population zone (LPZ) TEDE values (Ref. 10.1, RGP 4.2.2).

4.8.3 The occupancy and breathing rate of the maximum exposed individual presents in the control room are incorporated in design inputs 5.6.12 & 5.6.13 (Ref. 10.1, RGP 4.2.6).

4.8.4 10 CFR 50.67 (Ref. 10.4) establishes the following radiological criterion for the control room. This criterion is stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA (Ref. 10.1, RGP 4.4).

CR Dose Acceptance Criteria: 5 Rem TEDE (50.67(b)(2)(iii))

4.8.5 Credit for engineered safety features that mitigate airborne activity within the control room is taken for control room isolation or pressurization, intake or recirculation filtration (Ref. 10.1, RGP 4.2.4). The control room design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safety feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs. Several aspects of RMs can delay the isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response. The CR emergency filtration system is conservatively assumed to isolate and initiate at 30 minutes after a LOCA per Design Input 5.6.5.

4.8.6 The CR unfiltered in leakage is conservatively assumed to be 500 cfm (Design Input 5.6.7) during the CFREF transition period of 30 minutes after a LOCA. A conservative model would consider the normal ventilation mode for the transition period, which is of short duration (less than two minutes) until the control room envelop is fully pressurized following CREF initiation. Such a model would result in total unfiltered inleakage of 6,600 ft^3 (3000 ft^3/min x 2 min 1.1 = 6,600 ft^3). The conservative assumption of 500 cfm unfiltered inleakage during the transition period would result in 15,000 ft^3 (500 ft^3/min x 30 min = 15,000 ft^3) unfiltered air, which is 2 times higher.

4.8.7 No credits for KI pills or respirators are taken (Ref. 10.1, RGP 4.2.5).

5.0 DESIGN INPUTS:

5.1 General Considerations

5.1.1 Applicability of Prior Licensing Basis

The implementation of an AST is a significant change to the design basis of the facility and assumptions and design inputs used in the analyses. The characteristics of the AST and the revised TEDE dose calculation methodology may be incompatible with many of the analysis assumptions and methods currently used in the facility's design basis analyses. The HCGS plant specific design inputs and assumptions used in the TID-14844 analyses were assessed for their validity to represent the as-built condition of the plant and evaluated for their compatibility to meet the AST and TEDE methodology. The analysis in this calculation ensures that analysis assumptions, design inputs, and methods are compatible with the requirements of the AST and the TEDE criteria.

5.1.2 Credit for Engineered Safety Features

Credit is taken only for those accident mitigation features that are classified as safety-related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure modeled in this calculation is an 'A' or 'B' EDG failure concurrent with a loss of offsite power (LOP) resulting in the MSIV release at the ground level instead of released through the south plant vent (SPV). The consequences of an EDG failure is translated throughout the calculation by assuming that only four out of six FRVS recirculation filtration trains are available and one out of four inboard MSIV fails open. Assumptions regarding the occurrence and timing of a LOP are selected for the CREF system with the objective of maximizing the postulated radiological consequences.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to analyses required by 10 CFR 50.67 are compatible to AST and TEDE dose criteria and selected with the objective of maximizing the postulated dose. As a conservative alternative, the limiting value applicable to each portion of the analysis is used in the evaluation of that portion. The use of containment, ESF, and MSIV leakage values higher than actually measured, use of 10% lower flow rates for the FRVS and CREFS recirculation systems, use of 10% higher flow rate for FRVS vent, 30 minutes delay in the CREF initiation time, and use of ground release χ/Qs demonstrate the inherent conservatisms in the plant design and post-accident response. Most of the design input parameter values used in the analysis are those specified in the Technical Specifications (Ref. 10.6).

5.1.4 Meteorology Considerations

Atmospheric dispersion factors (χ/Qs) for the onsite release points such as the FRVS vent for containment and ESF leakage release path and turbine building louvers for MSIV leakage release path are re-developed (Ref. 10.5) using the NRC sponsored computer code ARCON96 and guidance provided in Draft NEI 99-03, Appendix D (Ref. 10.34). The EAB and LPZ χ/Qs are reconstituted using the HCGS plant specific meteorology and appropriate regulatory guidance (Ref. 10.32). The site boundary χ/Qs reconstituted in Reference 10.32 were accepted by the staff in the previous licensing proceedings.

5.2 Accident-Specific Design Inputs/Assumptions

The design inputs/assumptions utilized in the EAB, LPZ, and CR habitability analyses are listed in the following sections. The design inputs are compatible with the requirements of the AST and TEDE dose criteria and the assumptions are consistent with those identified in Regulatory Position 3 and Appendix A of RG 1.183 (Ref. 10.1). The design inputs and assumptions in the following sections represent the as-built design of the plant.

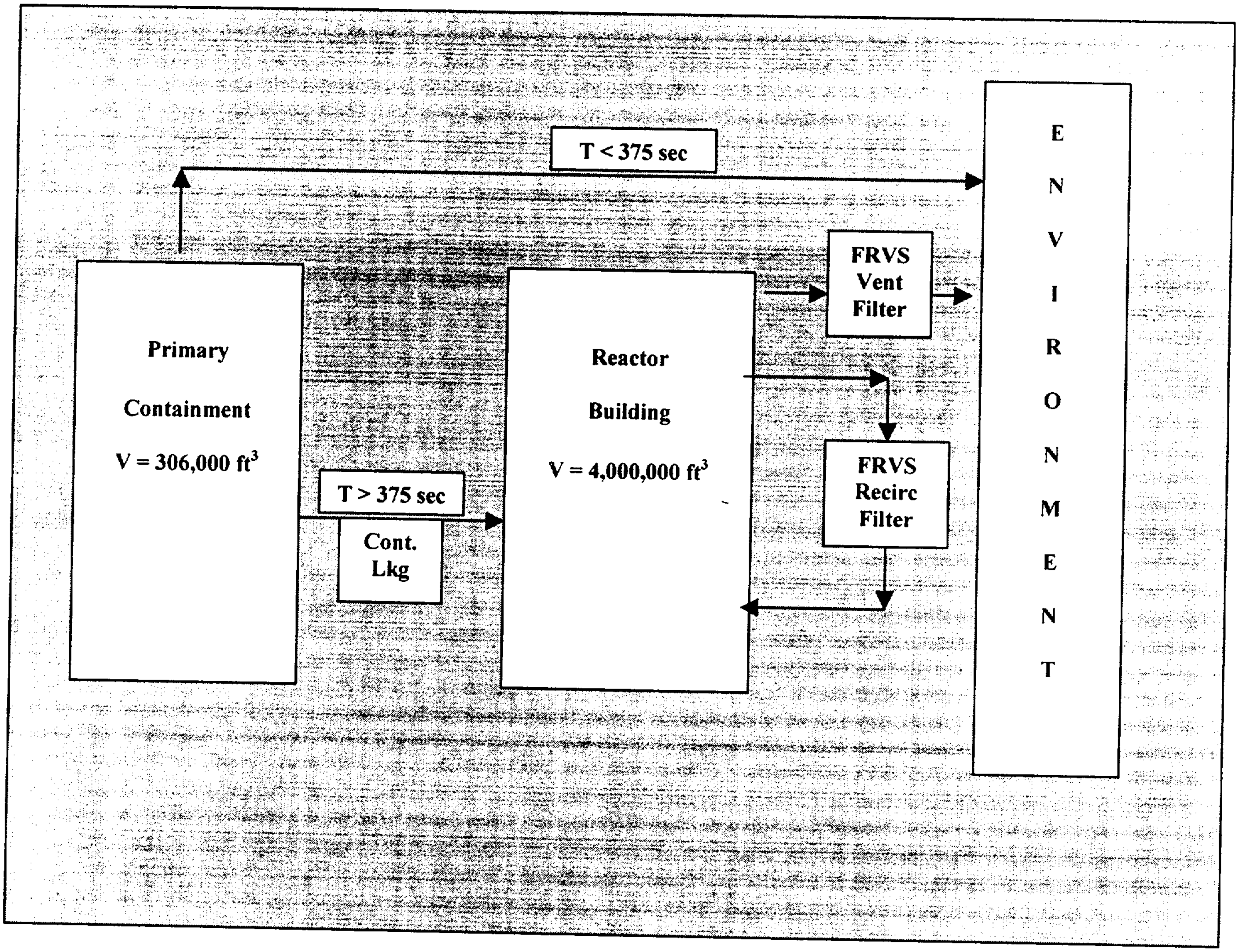


Figure 1: Containment Leakage RADTRAD Nodalization

Design Input Parameter	Value Assigned	Reference
5.3 Containment Leakage Model Parameters		
5.3.1 Source Term		
5.3.1.1 Power Level	3293 x 1.05 = 3458 MWt	10.6.9
5.3.1.2 Post-LOCA Containment Condition (Ref. 10.15)		
0-0.5 hr (Cont. Pressure)	63 psia	10.15
0.5- 720 hr (Cont. Pressure)	31 psia	
5.3.1.3 Isotopic Core Inventory (Curie) (Ref. 10.45) See Below		

Table 1

Isotope	Activity	Isotope	Activity	Isotope	Activity
CO-58	5.287E+05	RU103	1.503E+08	CS136	5.122E+06
CO-60	6.328E+05	RU105	1.071E+08	CS137	1.363E+07
KR 85	1.157E+06	RU106	6.074E+07	BA139	1.745E+08
KR 85M	2.788E+07	RH105	9.970E+07	BA140	1.677E+08
RB 86	1.743E+05	SB127	1.034E+07	LA140	1.728E+08
KR 87	5.454E+07	SB129	3.080E+07	LA141	1.594E+08
KR 88	7.691E+07	TE127M	1.359E+06	LA142	1.554E+08
SR 89	9.386E+07	TE127	1.024E+07	CE141	1.573E+08
SR 90	9.213E+06	TE129M	4.517E+06	CE143	1.513E+08
SR 91	1.274E+08	TE129	3.030E+07	CE144	1.178E+08
SR 92	1.352E+08	TE131M	1.383E+07	PR143	1.456E+08
Y 90	9.555E+06	TE132	1.333E+08	ND147	6.294E+07
Y 91	1.184E+08	I131	9.406E+07	NP239	2.050E+09
Y 92	1.357E+08	I132	1.356E+08	PU238	3.658E+05
Y 93	1.533E+08	I133	1.917E+08	PU239	3.890E+04
ZR 95	1.566E+08	I134	2.122E+08	PU240	4.995E+04
ZR 97	1.599E+08	I135	1.792E+08	PU241	1.774E+07
NB 95	1.561E+08	XE133	1.869E+08	AM241	2.455E+04
MO 99	1.739E+08	XE135	5.420E+07	CM242	7.032E+06
TC 99M	1.522E+08	CS134	1.869E+07	CM244	5.764E+05

Design Input Parameter	Value Assigned	Reference
5.3.1.4 Radionuclide Composition		

Table 2

Group	Elements	
Noble Gases	Xe, Kr	10.1, RGP 3.4, Table 5
Halogens	I, Br	
Alkali Metals	Cs, Rb	
Tellurium Group	Te, Sb, Se, Ba, Sr	
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co	
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	
Cerium	Ce, Pu, Np	

5.3.1.5 Release Fraction (Ref 10.1, Tables 1)

Table 3

BWR Core Inventory Fraction Released Into Containment

4.9 Group	Gap Release Phase	Early In-Vessel Release Phase
Noble Gases	0.05	0.95
Halogens	0.05	0.25
Alkali Metals	0.05	0.20
Tellurium Metals	0.00	0.05
Ba, Sr	0.00	0.02
Noble Metals	0.00	0.0025
Cerium Group	0.00	0.0005
Lanthanides	0.00	0.0002

5.3.1.6 Timing of Release Phase (Ref. 10.1, Table 4)

Table 4

Phase	Onset	Duration
Gap Release	2-min	0.5 hr
Early In-Vessel Release	0.5 hr	1.5 hr

Design Input Parameter	Value Assigned	Reference
5.3.1.7 Iodine Chemical Form		
	Table 5	
Iodine Chemical Form	**%**	
Aerosol	95.0%	10.1, RGP 3.5
Elemental	4.85%	
Organic	0.15%	
5.3.1.8 Post-LOCA Drywell Temperature		
	Table 6	
Post-LOCA Time (Hr)	Temperature (^{0}F)	
0	340	Temperature values are bounding based on information in Reference 10.25, pages 35 through 45.
3	320	
6	250	
24	208	
96	180	
240	170	
480	150	
720		
5.3.2 Activity Transport in Primary Containment		
5.3.2.1 Primary Containment Parameters		
5.3.2.2 Drywell Air Volume	169000 ft^3	10.6.6 & 10.16
5.3.2.3 Suppression Chamber Air Volume	137000 ft^3	10.6.6 & 10.16
5.3.2.4 Containment Air Volume	306000 ft^3	DI 5.3.2.2 + DI 5.3.2.3
5.3.2.5 Containment Leak Rate		
0-24 hrs	0.5 v%/day	10.6.4 & 10.15
24-720 hrs	0.25 v%/day	10.1, RGP A.3.7 & 10.15
5.3.2.6 Draw Down Time	375 sec	10.6.8
5.3.2.7 Cont. Leakage Before Draw Down Time (< 375 sec)	Directly Released to Environment	10.1, RGP A.4.2
5.3.2.8 Cont. Leakage After Draw Down Time (>375 sec)	Directly Released to Reactor Building	10.1, RGP A.4.2

Design Input Parameter	Value Assigned	Reference
5.3.2.9 Reactor Building Volume	4,000,000 ft^3	10.6.7
5.3.2.10 Reactor Building Mixing	50%	10.1, RGP A.4.4
5.3.2.11 FRVS Vent Exhaust Rate Before Draw Down	9000 cfm ± 10%	10.6.3, & 10.20
5.3.2.12 FRVS Vent Exhaust Flow Rate After Draw Down	$3324 + 5676e^{-1.18t}$	Actual Eqn in Ref. 10.19, page 24 is $3324 + 5637e^{-1.18t}$
5.3.2.13 FRVS Vent Exhaust Filter Efficiency		

Table 7

Iodine Species	Efficiency (%)	
Elemental	99%	10.6.2 & 10.10, Table 2
Aerosol	99%	10.6.1 & 10.10, Table 2
Organic	99%	10.6.2 & 10.10, Table 2

5.3.2.14 Post Draw Down FRVS Exhaust Rates For 50% Mixing (using Design Input 5.3.2.12)

Table 8

Post-LOCA Time (hr)	Normal Flow Rate (cfm) $A = 3324 + 5676e^{-1.18t}$	50% Mixing Flow Rate (cfm) A x 1.1 x 2
0	9000	19800
0.1042 (375 sec)	8343	18355
0.3333	7154	15739
2	3860	8492
4	3375	7425
8	3324	7313
24	3324	7313
96	3324	7313

5.3.2.15 FRVS Recirc Flow Rate	120000 cfm - 10% (or, 108,000 cfm)	10.6.12 &10.20
5.3.2.16 FRVS Recirc Filter Efficiency		

Table 9

Iodine Species	Efficiency (%)	
Elemental	80%	10.6.11
Aerosol	99%	10.6.10
Organic	80%	10.6.11

CALC. NO.: H-1-ZZ-MDC-1880			REFERENCE:			
ORIGINATOR, DATE	REV:	G. Patel, 05/15/01	0			
REVIEWER/VERIFIER, DATE		J. Duffy, 5/16/01				

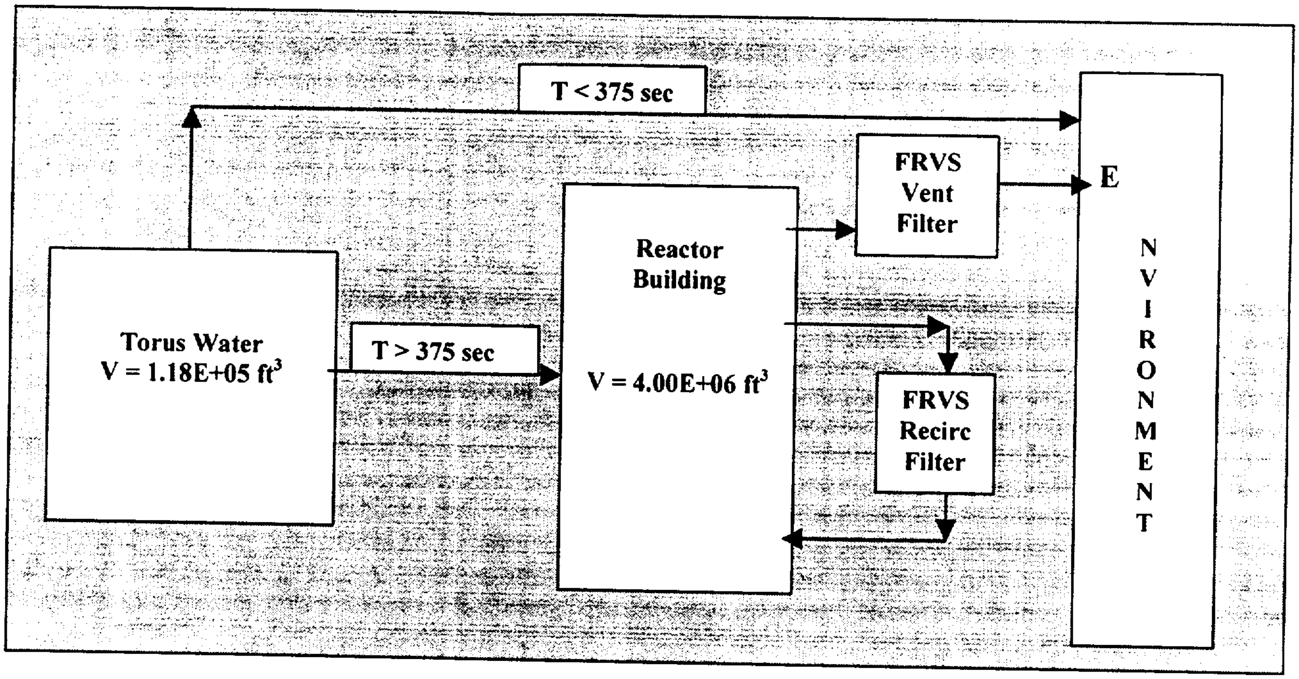


Figure 2: HCGS ESF Leakage RADTRAD Nodalization

Design Input Parameter	Value Assigned	Reference
5.4 ESF Leakage Model Parameters		
5.4.1 Sump Water Volume	118,000 ft^3	10.6.5 & 10.16
5.4.2 ESF Leakage	10 gpm	10.18, page 13
5.4.3 ESF Leakage Initiation Time	0 minute	Assumption
5.4.4 Suppression Pool Water pH	>7	10.1, RGP A.2
5.4.5 Sump Water Activity (Ref. 10.1, RGP A.5.1, A.5.3 & Tables 1 & 4)		
	Table 10	
Group	Gap Release Phase	Early In-Vessel Release Phase
Timing Duration (Hrs)	2 min – 0.50 Hr	0.50 – 2.0 Hr
Halogen	0.05	0.25
5.4.6 Iodine Flashing Factor	10%	10.1, RGP A.5.5 and 10.25, page 35 through 45
5.4.7 Chemical Form Iodine In ESF Leakage		
Elemental	97%	10.1, RGP A.5.6
Organic	3%	

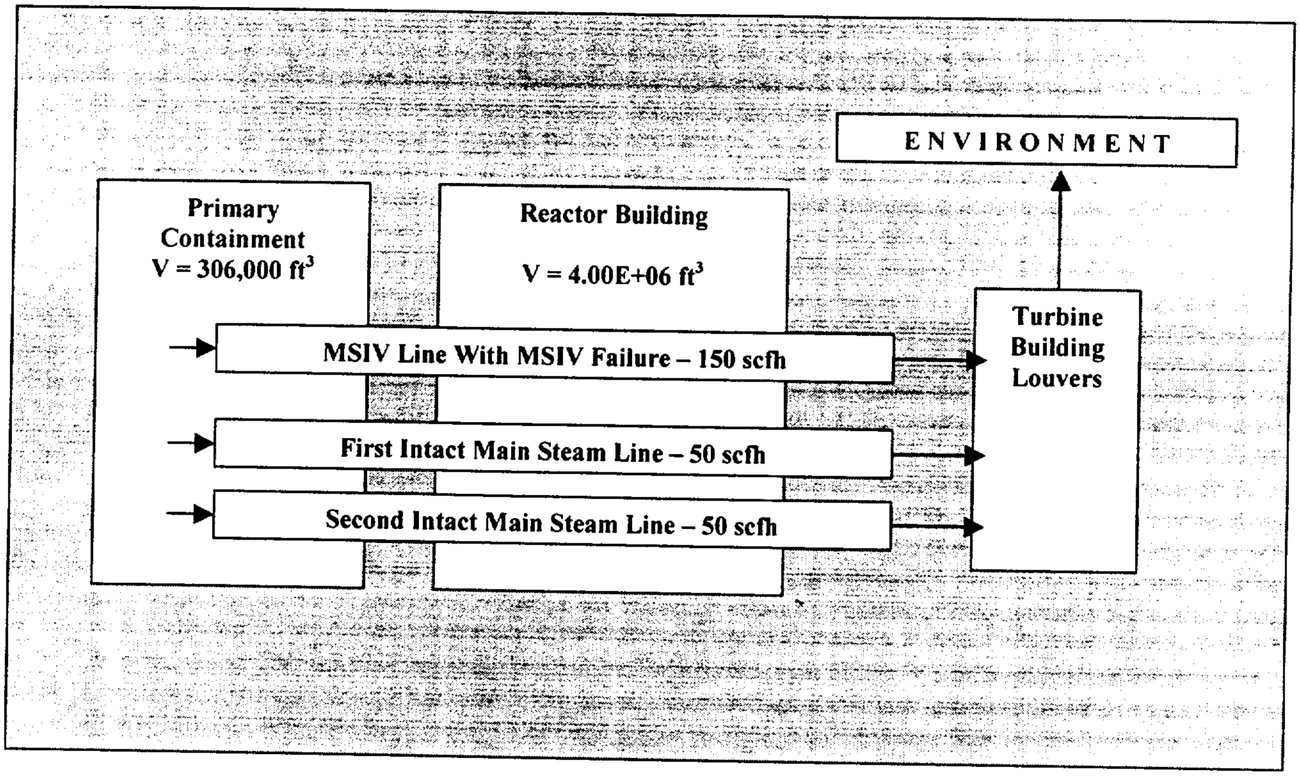


Figure 3: HCGS MSIV Leak RADTRAD Nodalization

Design Input Parameter	Value Assigned	Reference
5.5 MSIV Leakage Model Parameters		
5.5.1 Total MSIV Leak Rate Thru All Four Lines	≤ 250 scfh	Proposed Limit to TS 3.6.1.2.c
5.5.2 MSIV Leak Rate Through Line With MSIV Failed	150 scfh	Assumed
5.5.3 MSIV Leak Rate Through First Intact Line	50 scfh	Assumed
5.5.4 MSIV Leak Rate Through Second Intact Line	50 scfh	Assumed
5.5.5 Number of Steam Lines	4	10.11 & 10.12e
5.5.6 Diameter and Wall Thickness of Pipe Between RPV Nozzle & Inboard Isolation Valves HV F022A/B/C/D	Diameter = 26" Wall Thickness = 1.117"	10.13b 10.14c

Design Input Parameter	Value Assigned	Reference
5.5.7 Diameter and Wall Thickness of Pipe Between Inboard & Outboard Isolation Valves HV F028A/B/C/D	Diameter = 26" Wall Thickness = 1.117"	10.12e 10.14c
5.5.8 Diameter and Wall Thickness of Pipe Between Outboard & 3rd Isolation Valves HV 3631A/B/C/D	Diameter = 26" Wall Thickness = 1.023	10.12e 10.14a
5.5.9 Diameter of Pipe Between 3rd Isolation & Turbine Stop Valves MSV1/2/3/4	Diameter = 28" Wall Thickness = 0.934"	10.12a 10.14b
5.5.10 Corrosion Allowance For Steam	0.12"	10.14

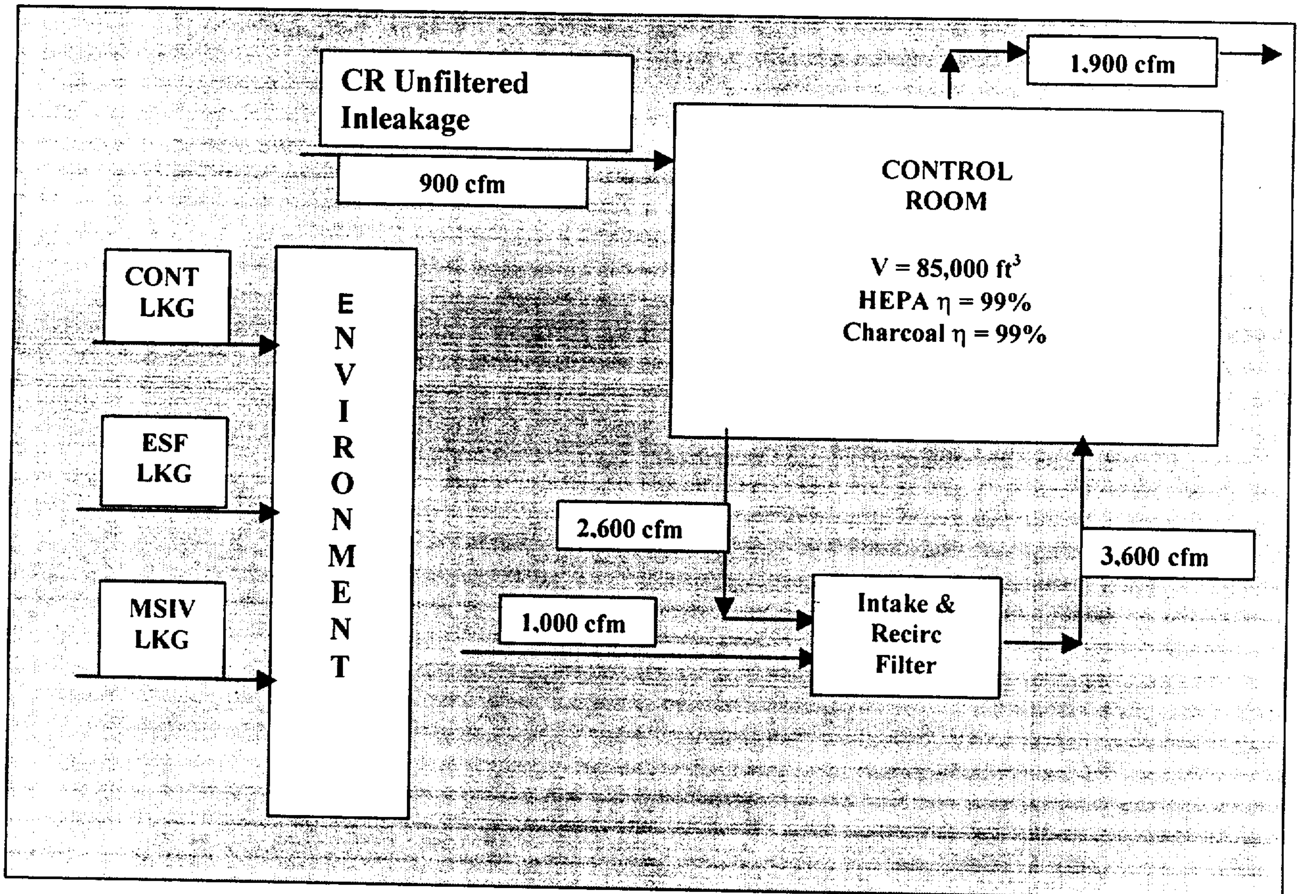


Figure 4 – HCGS Control Room RADTRAD Nodalization

Design Input Parameter	Value Assigned	Reference
5.6 Control Room Model Parameters		
5.6.1 CR Volume	85,000 ft^3	10.33, Page 10
5.6.2 CREF System Flow Rate	1,000 cfm	10.6.16
5.6.3 CR Minimum Recir Flow Rate	2,600 cfm	10.6.15
5.6.4 CR Unfiltered Inleakage	900 cfm	Assumed
5.6.5 CREV System Initiation Time After a LOCA	30 minutes	Assumption 4.8.5
5.6.6 CR Charcoal & HEPA Filter Efficiencies	99%	10.6.13 & 10.6.14

Design Input Parameter	Value Assigned	Reference
5.6.7 CR Unfiltered Inleakage During Pressurization	500 cfm	10.40, page 6.4-8 & Assumption 4.8.6
5.6.8 CR Concrete Wall, Floor, and Ceiling Thickness		
Walls	>3 feet	10.27 through 10.31
Floor	>3 feet	
Total Roof Thickness	2'-10-1/2"	
Ceiling Above CR	1'-0"	10.29a & 10.29b
5.6.9 CR χ/Qs For Containment & ESF Leakage Release Via FRVS Vent Ground Level Release		
	Table 11	
Time	**X/Q (sec/m^3)**	
0-2	1.26E-03	10.5, page 23
2-8	8.25E-04	
8-24	3.35E-04	
24-96	2.39E-04	
96-720	1.76E-04	
5.6.10 CR X/Qs For MSIV Leakage Release Via Turbine Building Louvers Ground Level Release		
	Table 12	
Time	**X/Q (sec/m^3)**	10.5, page 24
0-2	6.00E-04	
2-8	3.93E-04	
8-24	1.49E-04	
24-96	1.00E-04	
96-720	7.66E-05	
5.6.11 CR Occupancy Factors		
	Table 13	
Time (Hr)	%	
0-24	100	10.1, RGP 4.2.6
24-96	60	
96-720	40	

Design Input Parameter	Value Assigned	Reference
5.6.12 CR Breathing Rate	3.5E-04 (m^3/sec)	10.1, RGP 4.2.6
5.6.13 Minimum Reactor Bldg Wall Thickness	1'-6"	10.35
5.7 Site Boundary Release Model Parameters		
5.7.1 EAB X/Q (0-2 Hrs)	1.9E-04 sec/m^3	10.32, pages 5 & 9
5.7.2 LPZ X/Qs (0-720 Hrs)		

Table 14

Time	X/Q (sec/m^3)	
0-2	1.9E-05	10.32, pages 5 & 9
2-4	1.2E-05	
4-8	8.0E-6	
8-24	4.0E-06	
24-96	1.7E-06	
96-720	4.7E-07	

5.7.3 Offsite Breathing Rate

Table 15

Time	(m^3/sec)	
0-8	3.5E-04	10.1, RGPs 4.1.3 & 4.4
8-24	1.8E-04	
24-720	2.3E-04	

5.7.4 CR Charcoal Filter Dimensions Approximated Conservatively		
5.7.4.1 Length	3 feet	10.38
5.7.4.2 Height	3 feet	
5.7.4.3 Width	4 feet	
5.7.5 Charcoal Density	0.70 g/cc	Assumed
5.7.6 Concrete Density	2.3 g/cc	Assumed
5.7.7 Dose Point Location	143'-0"	6' above EL 137'-0"

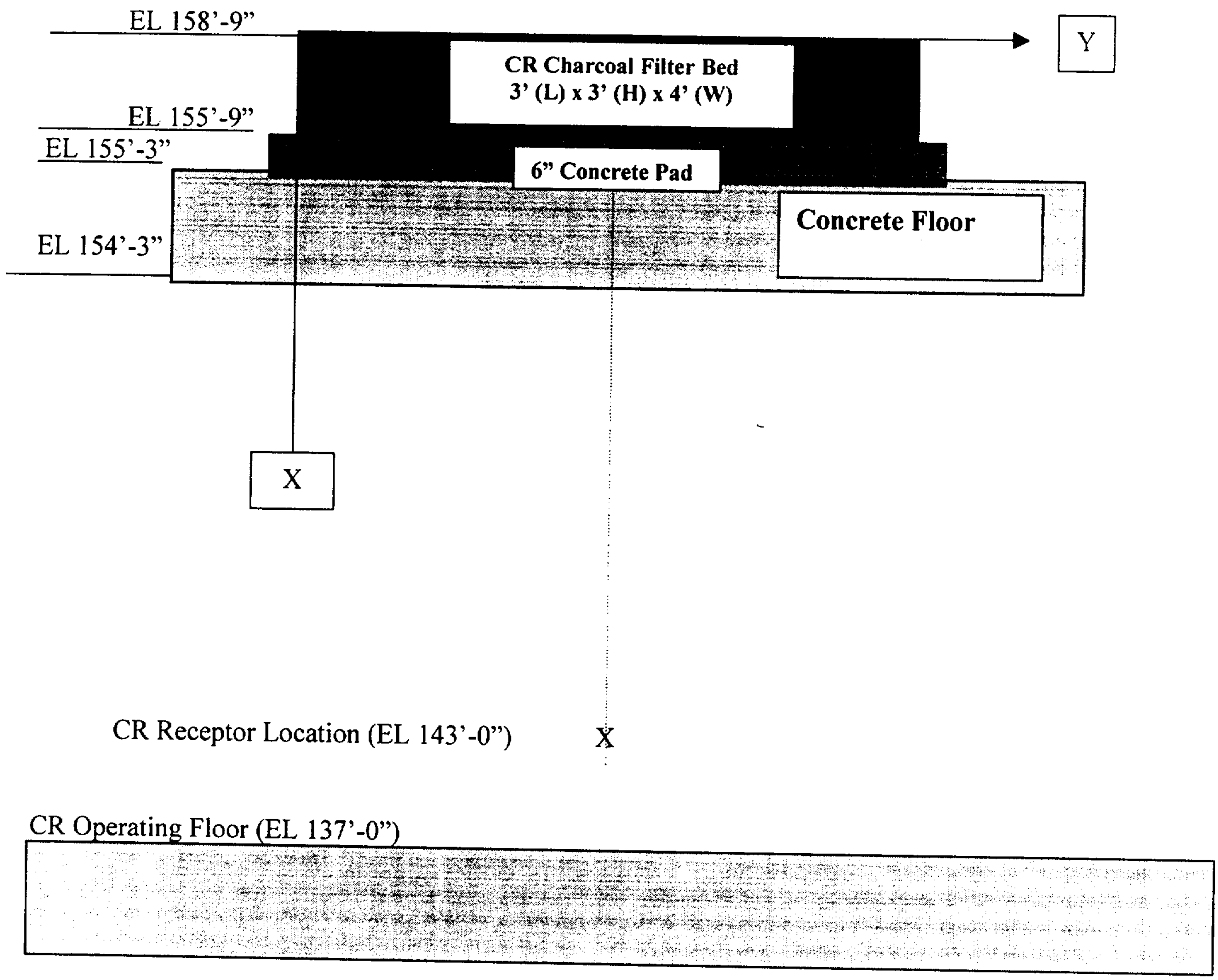
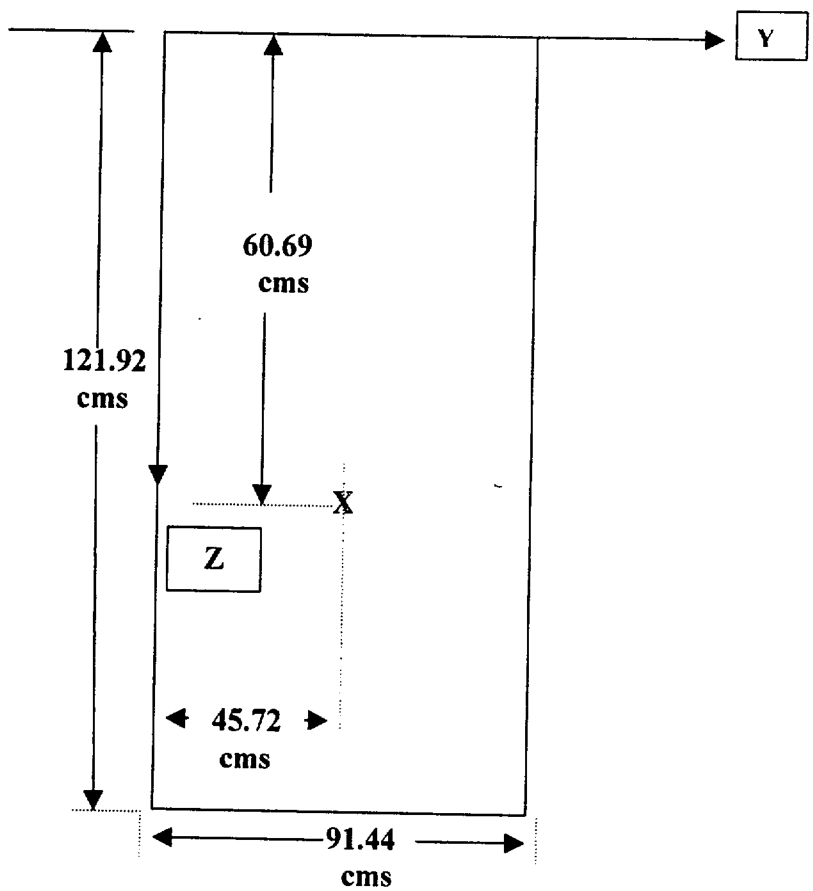


Figure 5 – CR Filter Shine Dose
(Elevation View)



X Indicates Dose Point Location

Figure 6 – CR Filter Shine Dose
(Plan View)

6.0 METHODOLOGY

The design basis accidents postulated were analyzed using a conservative set of assumptions and as-built design inputs to demonstrate the performance of one or more aspects of the facility design to protect the control room operator and the health and safety of the general public. The guidance in the Regulatory Guide 1.183 (Ref. 10.1) is followed line by line along with the plant-specific design input parameters computable for the AST and TEDE dose criteria. The numeric values of the post-accident performance of ESF components are conservatively selected to assure an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Any deviations from the methodology of Regulatory Guide 1.183 were only performed for plant specific situations if an adequate justification/bases exist.

6.1 Post-LOCA Containment Leakage:

6.1.1 Source Term:

The post-LOCA containment leakage model is shown in Figure 1. The core inventory listed in the Table 1 above is released into the containment at the release timing and fractions shown in Tables 3 & 4 (Ref. 10.1, RGPs 3.2 & 3.3). Since the post-LOCA minimum suppression chamber water pH is calculated at a value greater than 7.0 (Ref. 10.43), the chemical form of radioiodine released into the containment is assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide as shown in Table 5. With the exception of elemental and organic iodine and noble gases, the remaining fission products are assumed to be in particulate form (Ref 10.1, RGP 3.5). The RADTRAD plant-specific Nuclide Inventory File (NIF) is shown in Table 16. The isotopic Ci/MW_t is calculated in Table 16 and the RADTRAD NIF HCGSMHA_DEFis shown in Attachment A and used for the containment, ESF, and MSIV leakage paths. The source term design inputs are shown in Sections 5.3.1.1 through 5.3.1.8.

6.1.2 Transport In Primary Containment:

The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment as it is released. The radioactivity release into the containment is assumed to terminate at the end of the early in-vessel phase, which occurs at the end of 2 hrs after the onset of a LOCA (see Table 4). The design inputs for the transport in the primary containment are shown in Sections 5.3.2.1 through 5.3.2.9.

6.1.3 Reduction In Airborne Activity Inside Containment

The airborne iodine and aerosol are removed from the reactor building environment by the FRVS recirculation system, which re-circulates air at a design rate of 108,000 cfm or 1.62 vol/hr (108,000

ft^3/min x 60 min/hr x (4.00E+06 ft^3)$^{-1}$ = 1.62 vol/hr). Although, the FRVS recirc provides a good mixing of activity in the reactor building (RB), the airborne activity is conservatively assumed to mix with only 50% of the RB volume (Ref. 10.1, RGP A.4.4.). To simulate the 50% mixing in the RB, the exhaust rate of FRVS vent system is doubled as shown in Design Input 5.3.2.14. The FRVS vent exhaust rate varies with time as shown by the equation in Design Input 5.3.2.12. Table 8 provides the FRVS exhaust flow rates at 100% and 50% mixings. The airborne activity in the RB is removed by both the FRVS recirculation and FRVS vent filtration system before it released to environment. The charcoal and HEPA filtration efficiencies are shown in Section 5.3.2.16.

6.1.4 Dual Containment:

Leakage from the primary containment is assumed to be released directly to the environment prior to draw down time during which the RB does not maintain a negative pressure as defined in technical specifications (Ref 10.1, RGP A.4.2). 50% mixing is credited for dilution of the activity in the RB (Ref. 10.1, RGP A.4.4). The containment leakage RADTRAD input and output files are listed in the Attachments B and C and the EAB, LPZ, and CR TEDE doses are shown in the Section 8.0.

6.1.5 Containment Purging:

The HCGS containment is not purged for combustible gas or pressure control measure within 30 days of the LOCA. Therefore, the release containment purging is not analyzed per RG 1.183, RGP A.7.

6.2 Post-LOCA ESF Leakage:

The post-LOCA ESF leakage release model is shown in Figure 2. The ESF systems that recirculate suppression pool water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands; pump shaft seals, flanged connections, and other similar components. The radiological consequences from the postulated leakage is analyzed and combined with consequences from other fission product release paths to determine the total calculated radiological consequences from the LOCA (see Section 8.0 of this calc). The ESF components are located in the RB.

6.2.1 Source Term:

With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Sections 5.3.1.3 & 5.3.1.5) are assumed to instantaneously and homogeneously mix in the suppression pool water at the time of release from the core. The total ESF leakage from all components in the ESF recirculation systems is 10 gpm. This ESF leakage is doubled (Ref 10.1, RGP A.5.2) and assumed to start at time t=0.0 minute after onset of a LOCA. With the exception of iodine, all remaining fission products in the recirculating liquid are assumed to be retained in the liquid phase. The design inputs for the ESF leakage are shown in Sections 5.4.1 through 5.4.6.

6.2.3 Chemical Form

The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic (Ref. 10.1, RGP A.5.6) based on the Regulatory Position A.5.6. The reduction in ESF leakage activity by dilution in the RB and removal by FRVS recirc and FRVS vent filtration systems are credited.

The ESF leakage RADTRAD inputs and outputs files are listed in the Attachments D & F and the EAB, LPZ, and CR TEDE doses are shown in the Section 8.0.

6.3 Post-LOCA MSIV Leakage:

The main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage are analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.

6.3.1 Source Term

For the purpose of this analysis, the activity available for release via MSIV leakage is assumed to be that activity released in the drywell for evaluating containment leakage.

A total of 250 scfh (the maximum proposed allowable leakage limit) is assumed to occur as follows:

(1) 150 scfh through the steam line with the failed MSIV. The plate out of activity and holdup time are not credits in the steam line between the inboard and outboard valves. The plateout and holdup are conservatively credited in the steam lines from the RPV to inboard isolation valve, outboard isolation valve to turbine block valve.

(2) 50 scfh through a first intact steam line. The plate out of activity and holdup time are credits in the entire steam line from the RPV nozzle to turbine stop valve.

(3) 50 scfh through a second intact steam line. The plate out of activity and holdup time are credits in the entire steam line from the RPV nozzle to turbine stop valve.

The MSIV leakage is assumed to continue for entire duration of the accident. Per RG 1.183, RGP A.6.2 (Ref. 10.1), the MSIV leakage is to reduce to a value 50% of the maximum leak rate after the first 24 hours, based on the post-LOCA drywell pressure (Ref. 10.15).

Reduction of the amount of released aerosol radioactivity by gravitational deposition on the pipe surface is calculated in Section 7.4 using the NRC staff Monte Carlo analysis to determine the distribution of settling velocity in well mixed flow in the steam line (Ref 10.22). The analysis in Section 7.4.1 takes the credit of pipe surface areas upstream and down stream of inboard isolation valves because the steam

lines from the RPV nozzle to turbine stop valve are seismically designed and supported for Safe Shutdown Earthquake (SSE) (Ref 10.26 & 10.37). The analysis in Section 7.4.1 determines that all airborne aerosol (100%) in MSIV leakage will be deposited on the steam pipe surface.

The reduction in elemental iodine activity in the MSIV leakage is calculated in Section 7.4.2 using the staff recommended guidance on acceptable method in Reference 10.23 (Reference A-9 of RG 1.183). Both, the temperature dependent elemental iodine deposition and resuspension rates, net iodine deposition rates, and iodine removal efficiencies are calculated in Tables 18 through 24 using J.E. Cline method (Ref 10.23). The remaining airborne activity in the MSIV leakage after removal by deposition (aerosol) and plateout (elemental) mechanisms is directly released to the environment as a ground level release through the turbine building louvers.

The holdup times for each MSIV leakage release path (MSIV failed and intact steam lines) are calculated in Sections 7.2 and 7.3 based on the leakage rates and well-mixed steam piping volumes. These parameters calculated in Sections 7.2, 7.3, & 7.4 are input in the RADTRAD MSIV release model to calculate EAB, LPZ, and CR doses, which are listed in Section 8.0. The design inputs for the MSIV leakage are shown in Sections 5.5.1 through 5.5.8.

6.4 Control Room Model

The post-LOCA control room RADTRAD nodalization is shown in Figure 4 with the design input parameters. The post-LOCA radioactive releases that contribute the CR TEDE dose are as follows:

- Post-LOCA Containment Leakage
- Post-LOCA ESF Leakage
- Post-LOCA MSIV Leakage

The radioactivity from the above sources are assumed to be released into the atmosphere and transported to the CR air intake, where it may leak into the CR envelope or be filtered by the CR intake and recirculation filtration system and distributed in the CR envelope. There are four major radioactive sources, which contribute to the CR TEDE dose are:

- Post-LOCA airborne activity inside the CR
- Post-LOCA airborne cloud external to CR
- Post-LOCA containment shine to CR
- Post-LOCA CREF filter shine

6.4.1 Post-LOCA Airborne Activity Inside CR

The post-LOCA radioactive releases from various sources are discussed in Sections 6.1 through 6.3 above and shown in Figure 4. The activities releases from the various sources are diluted by the atmospheric dispersions and carried to the CR air intake. The atmospheric dispersion factors are shown in Sections 5.6.9 & 5.6.10 for the containment/ESF and MSIV leakages. The containment and ESF leakages have the same release point (FRVS vent) and X/Qs. The RADTRAD release models are developed for each release path using appropriate design inputs from Sections 5.3, 5.4, and 5.5. The CR dose model is developed using the design input parameters in Sections 5.6.1 through 5.6.13. The CR airborne TEDE dose contributions from the above post-LOCA sources are calculated and tabulated in Section 8.0.

6.4.2 Post-LOCA Airborne Cloud External to CR

The radioactive plumes released from various post-LOCA sources are carried over the CR building, submerging the CR in the radioactive cloud. The CR operator is exposed to direct radiation from the radioactive cloud external to the CR structure. The review of control building concrete structure drawings (Ref. 10.27 through 10.31) indicate that the CR is surrounded by at least 2'-10-1/2" (1' ceiling @ EL 155'-3" and 1'-10-1/2" roof @ EL 172'0") concrete shielding with a minimum distance of 29 feet from the least shielding (172'-0" – (137'-0" + 6'-0" tall person)). This minimum-shielding configuration provides an adequate protection to the CR operator to reduce the CR operator external cloud dose to negligible amount.

6.4.3 Post-LOCA Containment Shine to CR

The post-LOCA airborne activity in the containment is released to reactor building via containment leakage through the penetrations and openings and gets uniformly distributed inside the RB. The airborne activity confined in the dome space of the RB contributes direct shine dose to CR operator. The review of the containment building concrete structure drawing (Ref. 10.35) indicates that the minimum dome concrete thickness is 1'-6". The CR minimum roof/ceiling concrete shielding is 2'-10-1/2". The combined concrete shielding of 4'-4-1/2" (1'-6" + 2'-10-1/2" = 4'-4-1/2") provides ample shielding to reduce the CR operator containment shine dose to an insignificant amount.

6.4.4 Post-LOCA CREF Filter Shine

The two trains of CREF charcoal and HEPA filters are located above the CR operating floor at elevation 155'-3" (Refs. 10.28, 10.29, & 10.39). The CR operating floor is located at elevation 137'-0" (Ref. 10.28c). The concrete floor at EL 155'-3" is 1 feet thick (Ref. 10.29). The filter assembly is placed on 6" concrete pad (Ref. 10.39c, Section DD), which provides the total concrete shielding of 1'-6" between the CR operator and charcoal/HEPA filter. The receptor location is assumed to be located at 6 feet above

the CR operating floor right below the center of charcoal filter. The iodine and aerosol activities are conservatively collected on the charcoal bed. The dimensions of charcoal filter housing are obtained from Reference 10.38 and conservatively approximated to 3' (L) x 3' (H) x 4' (W) as shown in Figure 5 with dose point location.

Post-LOCA Activity

The RADTRAD3.02 code does not provide the post-LOCA iodine and aerosol activities accumulated on charcoal and HEPA filters. Therefore the iodine and aerosol activities are conservatively calculated as follows:

1. The time dependent isotopic iodine and aerosol integrated activities in the CR due to the post-LOCA containment leakage are calculated without taking the credits for charcoal and HEPA filters (RADTRAD File HAST1000CL03.psf). The time dependent integrated activities are shown in Table 26.

2. The time dependent isotopic iodine and aerosol integrated activities in the CR due to the post-LOCA containment leakage are calculated with the credit of charcoal and HEPA filters credit (RADTRAD File HAST1000CL02.psf). The time dependent integrated activities are shown in Table 25.

3. The total isotopic iodine and aerosol activities on the CR filters due to the containment leakage are calculated in Table 27 (Case 1 – Case 2).

4. Similarly, the time dependent isotopic iodine and aerosol integrated activities in the CR due to the post-LOCA MSIV and ESF leakages are calculated in Tables 28 – 29 and Tables 30 – 31 respectively.

5. The total iodine and aerosol integrated activities on the CR filters are shown in Table 32.

6. The total isotopic activities are input into the MicroShield Computer code (Ref. 10.9) with the source geometry, dimension, and detector location to compute the direct dose rate from the CR filter. The direct dose from the CR filter shine is calculated in Section 7.6 using the CR occupancy factors.

6.5 CR & FRVS Charcoal/HEPA Filter Efficiencies

The CR, FRVS vent, and FRVS recirculation charcoal and HEPA filters are tested to comply with Generic Letter 99-02 requirements (Refs. 10.3 & 10.6). The in-place penetration testing acceptance requirements are given in Hope Creek Technical Specifications (Ref. 10.6). The filter efficiencies credited in this analysis are calculated in Section 7.7 based on the testing criteria in Reference 10.6 and GL 99-02 (Ref. 10.3).

7.0 CALCULATIONS

7.1 HCGS Plant Specific Nuclide Inventory File (NIF) For RADTRAD3.02 Input

The RADTRAD nuclide inventory file Bwr_def_NIF establishes the power dependent radionuclide activity in Ci/MW_t for the reactor core source term. Since these core radionuclide activities are dependent on the core thermal power level, reload design, and burnup, the NIF is modified based on the plant-specific core information. The Ci/MW_t for the core radionuclides are calculated in Table 14 below and the NIF for RADTRAD input is modified accordingly as shown in Appendix A.

7.2 Main Steam Line Volumes & Surface Area For Plateout of Activity

7.2.1 MSIV Line Between RPV Nozzle & Inboard Isolation Valve

Piping Class = DLA (Ref. 10.13b)

Pipe Diameter = 26" (Ref. 10.13b)

Minimum Wall Thickness = 1.117" (Ref. 10.14c)

Corrosion Allowance For Steam = 0.12" (Ref. 10.14c)

Total Minimum Thickness = 1.117" + 0.12" = 1.237"

26" Pipe ID = OD – (2 x Min Wall Thickness) = 26" – 2 x 1.237" = 23.526" = 1.961'

Shortest Length of Pipe Between RPV Nozzle & Inboard Isolation Valves ≡ 91 ' (Ref. 10.13)

Volume of Pipe Between RPV Nozzle & Inboard Isolation Valves

$= \pi r^2 L = \pi (1.961/2)^2 \times 91' = \boxed{274.84 \text{ ft}^3 = 7.79 \text{ m}^3}$

Surface Area $= \pi D L = \pi \times 1.961 \times 91 = \boxed{560.62 \text{ ft}^2 = 52.11 \text{ m}^2}$

7.2.2 MSIV Line Between Inboard & Outboard Isolation Valves

Piping Class = DLA (Ref. 10.13b)

Pipe Diameter = 26" (Ref. 10.13b)

Minimum Wall Thickness = 1.117" (Ref. 10.14c)

Corrosion Allowance For Steam = 0.12" (Ref. 10.14c)

Total Minimum Thickness = 1.117" + 0.12" = 1.237"

26" Pipe ID = OD – (2 x Min Wall Thickness) = 26" – 2 x 1.237" = 23.526" = 1.961'

Length of Pipe Between Inboard & Outboard Isolation Valves = 25.67' (Ref. 10.13)

Volume of Pipe Between Inboard & Outboard Isolation Valves

$= \pi r^2 L = \pi (1.961/2)^2 \times 25.67 =$ $\boxed{77.53 \text{ ft}^3 = 2.2 \text{ m}^3}$

Surface Area $= \pi D L = \pi \times 1.961 \times 25.67 =$ $\boxed{158.14 \text{ ft}^2 = 14.70 \text{ m}^2}$

7.2.3 <u>MSIV Line Between Outboard & Third Isolation Valves</u>

Piping Class = DBB (Ref. 10.11 & 10.12e)

Pipe Diameter = 26" (Ref. 10.12e)

Minimum Wall Thickness = 1.023" (Ref. 10.14a)

Corrosion Allowance For Steam = 0.12" (Ref. 10.14a)

Total Minimum Thickness = 1.023" + 0.12" = 1.143"

26" Pipe ID = OD – (2 x Min Wall Thickness) = 26" – 2 x 1.143" = 23.714" = 1.976'

Length of Pipe Between Outboard & Third Isolation Valves = 41.33' (Ref. 10.12e)

Volume of Pipe Between Outboard & Third Isolation Valves

$= \pi r^2 L = \pi (1.976/2)^2 \times 41.33' =$ $\boxed{126.74 \text{ ft}^3 = 3.59 \text{ m}^3}$

Surface Area $= \pi D L = \pi \times 1.976 \times 41.33 =$ $\boxed{256.57 \text{ ft}^2 = 23.85 \text{ m}^2}$

7.2.4 <u>MSIV Line Between Third Isolation Valve and Turbine Stop Valve</u>

Piping Class = DBC (Ref. 10.11 & 10.12a)

Pipe Diameter = 28" (Ref. 10.12a)

Minimum Wall Thickness = 0.934" (Ref. 10.14b)

Corrosion Allowance For Steam = 0.12" (Ref. 10.14b)

Total Minimum Thickness =0.934" + 0.12" = 1.054"

28" Pipe ID = OD – (2 x Min Wall Thickness) = 28" – 2 x 1.054" = 25.892" = 2.158'

Length of Pipe Between Third Isolation Valve & Turbine Stop Valve = 272.6' (Ref. 10.12a)

Volume of Pipe Between Third Isolation Valve & Turbine Stop Valve

$= \pi r^2 L = \pi (2.158/2)^2 \times 272.6' = \boxed{997.05 \text{ ft}^3 = 28.26 \text{ m}^3}$

Surface Area $= \pi D L = \pi \times 2.158 \text{ ft} \times 272.6 \text{ ft} = \boxed{1{,}848.11 \text{ ft}^2 = 171.78 \text{ m}^2}$

7.2.5 Surface Area & Volume of Failed MSIV Steam Line

Total Volume of MSIV Leakage Path For MSIV Failed Steam Line

$= 274.84 \text{ ft}^3 + 126.74 \text{ ft}^3 + 997.05 \text{ ft}^3$

$= \boxed{1{,}398.63 \text{ ft}^3 = 39.58 \text{ m}^3}$

Total Surface Area of MSIV Leakage Path For MSIV Failed Steam Line

$= 560.62 \text{ ft}^2 + 256.57 \text{ ft}^2 + 1{,}848.11 \text{ ft}^2$

$= \boxed{2{,}665.56 \text{ ft}^2 = 247.77 \text{ m}^2}$

7.2.6 Surface Area & Volume of Intact Steam Lines

Total Volume of MSIV Leakage Path For Intact Steam Lines

$= 274.84 \text{ ft}^3 + 77.53 \text{ ft}^3 + 126.74 \text{ ft}^3 + 997.05 \text{ ft}^3$

$= \boxed{1{,}476.16 \text{ ft}^3 = 41.83 \text{ m}^3}$

Total Surface Area of MSIV Leakage Path For Intact Steam Lines

$= 560.62 \text{ ft}^2 + 158.14 \text{ ft}^2 + 256.57 \text{ ft}^2 + 1{,}848.11 \text{ ft}^2$

$= \boxed{2{,}823.7 \text{ ft}^2 = 262.46 \text{ m}^2}$

7.3 Holdup Times

MSIV Leak Rate of 250 scfh
Holdup Time for MSIV Leakage of 150 scfh for each of two MSIV Failed Line

$= 1{,}398.63 \text{ ft}^3 / 150 \text{ ft}^3/\text{hr} = 9.32 \text{ hrs}$

Holdup Time for MSIV Leakage of 50 scfh/MSIV For MSIV Intact Lines

$= 1{,}476.16 \text{ ft}^3 / 50 \text{ ft}^3/\text{hr} = 29.52 \text{ hrs}$

MSIV Leak Rate For MSIV Failed Line = 150 $\text{ft}^3/\text{hr} \times 1/60$ hr/min = 2.50 cfm

MSIV Leak Rate For MSIV Intact Lines = 50 $\text{ft}^3/\text{hr} \times 1/60$ hr/min = 0.8334 cfm

MSIV leakage rate is halved after 24 hours

7.4 Plateout of Activity in Main Steam Lines

7.4.1 Aerosol Deposition

Reference 10.37 indicates that the HCGS main steam piping from the reactor pressure vessel (RPV) nozzle to the turbine stop valve is seismically analyzed to assure the piping wall integrity during and after a seismic event (SSE). The Hope Creek turbine building is classified as Non-seismic, however, codes and criteria similar to those for Seismic Category I structure, were used for the structure design of the entire building (Ref. 10.26, Section 1.2). The turbine building was dynamically analyzed and design to accommodate an SSE event (Ref. 10.37, page 1-2) so that it does not collapse on, or interact with, adjacent seismic Cat I structures for SSE. 10 CFR Part 100 requires that the structures, systems, and components necessary to ensure the capability of mitigating the radiological consequences of an accident that could result in exposures comparable to the does guideline of Part 100 be designed to remain functional during and following a safe-shutdown earthquake. The main steam lines and housing structures are qualified to meet Part 100 requirements; therefore, the main steam lines are credited for the aerosol deposition and holdup for MSIV leakage path.

The staff concludes that the plug flow is expected to result in less offsite release than well mixed flow, because the concentration of activity released to the environment is at the concentration of the material in the plug at the end of the pipe (Ref. 10.22, Appendix A, Page A-1). Plug flow effectively results in a longer fission product transport time in the pipe and more deposition in the pipe.

The staff utilized the following equations for the derivation of the well-mixed flow model (Ref. 10.22, page A-2 & A-3)

$$\lambda_s = \frac{\upsilon_s . A .}{V}$$

Where λ_s = rate constant for settling
υ_s = settling velocity
A = settling area
V = volume of well-mixed region

And $$\upsilon_s = \frac{\rho . d_e^2 . g . C_s}{18 . \mu . k}$$

Where ρ = aerosol density
d_e = aerosol diameter
g = gravitational acceleration
C_s = Cunningham slip factor
μ = viscosity

k = shape factor

The values of aerosol density, diameter, and shape factor during a design basis LOCA have some uncertainty. Therefore, the staff performed a Monte Carlo analysis to determine the distribution of settling velocities in the main steam line during the in-vessel release phase. The results of Monte Carlo analysis for settling velocity in the main steam line is as follows:

Percentile	Settling Velocity (m/sec)	Removal Rate Constant (hr^{-1})
60th (average)	0.00148	11.43
50th (median)	0.00117	9.04
40th	0.00081	6.26
10th	0.00021	1.62

The 40 percentile settling velocity is selected to calculate the aerosol removal rate and efficiency using the Hope Creek plant specific piping parameters.

Settling velocity $\upsilon_s = 0.00081$ m/sec
Settling area $A = 262.46\ m^2$
Main steam piping volume of well-mixed MSIV leakage $V = 41.83\ m^3$

$$\lambda_s = \frac{\upsilon_s . A}{V} = \frac{0.00081\ \text{m/sec} \times 262.46\ m^2}{41.83\ m^3} \times 3600\ \text{sec/hr} = 18.30\ hr^{-1}$$

Aerosol Removal Efficiency $= (1 - e^{-\lambda_s}) \times 100 = (1 - e^{-(18.30)}) \times 100 = (1 - (1.1E\text{-}08)) \times 100 = 100\%$

It means that all aerosols in the MSIV leakage will deposit on the large surface area of the main steam line. The aerosol being heavier in comparison to elemental iodine, it will get fixed on the surface. The settling velocity in the above analysis is a gravitational deposition velocity. Additional conservatisms include additional deposition by thermophoresis, diffusiophoresis, hydroscopicity, flow irregularities, and possible plugging of the leaking MSIV by aerosols.

7.4.2 Elemental Iodine

Gaseous iodine tends to deposit on the piping surface by chemical adsorption. The elemental iodine being the most reactive has the highest deposition rate. The iodine deposited on the surface undergoes both physical and chemical changes and can be re-emitted as an airborne gas (re-suspension) or permanently fixed to the surface (fixation). The RGP A.6.5 (Ref. 10.1) indicates that Reference A-9 provides acceptable models for deposition of iodine on the pipe surface. Reference 10.23, which is

Reference A-9 of Regulatory Guide 1.183 is used to determine the deposition and resuspension rates of elemental iodine as follows:

d_i = elemental iodine vapor deposition velocity (cm/s) = $\mathbf{e^{(2809/T\ -\ 12.80\ (\pm\ 0.33))}} = \mathbf{e^{(2809/T\ -\ 12.5)}}$ (Ref. 10.23, pages 4 & 12).

Where T = gas temperature (^{0}K)

This equation is same as equation 30 in Bixler Model in the RADTRAD3.02 code (Ref. 10.2, page 212).

The elemental iodine deposition velocities are calculated in Table 17 based the post-LOCA drywell temperature shown in Design Input 5.3.1.8.

The elemental iodine deposition rate λ_{ed} (hr^{-1}) = $\mathbf{\dfrac{d_i * S * 3600}{V}}$ (Ref. 10.23, page 4)

Where d_i = deposition velocity (m/sec)
S = surface area of deposition (m^2)
V = volume (m^3)

The deposition velocity in cm/sec, which is converted into m/sec and elemental iodine deposition rates at various drywell temperatures are calculated in Tables 18 & 19 for the MSIV failed and intact steam lines respectively.

The portion of elemental iodine deposited on the pipe surface will be resuspended as an airborne gas (organic iodine). Since the CR filtration efficiencies are same for all iodine spices, the resuspension of elemental iodine will produce the same thyroid organ dose irrespective of the form of iodine.

Resuspension rate of elemental iodine (sec^{-1})

$= \mathbf{2.32\ (\pm 2.00) \times 10^{-5}\ e^{-600/T} = 4.32 \times 10^{-5}\ e^{-600/T}}$

Resuspension rate of elemental iodine λ_{er} (hr^{-1})

$= \mathbf{4.32 \times 3600 \times 10^{-5}\ e^{-600/T}}$

The resuspension rates of elemental iodine at various drywell temperatures are calculated in Table 20.

The net deposition of elemental iodine on the pipe surface is the difference of deposition rate and resuspension rate. The net elemental iodine deposition rates at various drywell temperatures are calculated in Tables 21 and 22.

Net Deposition Rate of Elemental Iodine $\lambda_e = \lambda_{ed} - \lambda_{er}$

$1/DF = 1 - \eta = \exp^{(-\lambda e^* t)}$ (Ref 10.2, Equations 4 & 5, page 196)

Where DF = decontamination factor
η = filter efficiency for elemental iodine
λe = elemental iodine removal rate (hr^{-1})
t = time (hr)

Therefore, Elemental Iodine Filter Efficiency = **$1 - e^{-(\lambda e * t)}$**

The values net elemental iodine deposition rates (λe) are obtained from Table 20 and the corresponding filter efficiencies at various drywell temperatures are calculated in Tables 23 & 24 for the MSIV failed and intact steam lines respectively. The conservative values are used for each time step in RADTRAD model rather than using average values for each time step.

The elemental iodine removal efficiencies at various drywell temperatures are used along with aerosol removal efficiency (Section 7.4.1) in the RADTRAD3.02 MSIV release model.

7.5 ESF Leak Rates

The design basis ESF leakage is 10 gpm, which is doubled and converted into cfm as follows:

10 gallon/min x 2 x 1/7.481 ft^3/gallon = 2.673 cfm

10% of ESF leakage becomes airborne = 0.10 x 2.673 = 0.2673 cfm

7.6 CR Direct Dose From Filter Shine

CR Filter Shine Dose Rate = 7.754E-03 mRem/hr

CR Operator Exposure Time = 1 x (24 hr) + 0.60 (96 hr – 24 hr) + 0.40 (720 hr – 96 hr)

= 24 hr + 0.60 (72 hr) + 0.40 (624 hr) = 316.8 hr

Total CR Dose From Filter Shine

= 7.754E-03 mRem/hr x 1/1000 Rem/mRem x 316.8 hr = 2.4565E-03 Rem

7.7 FRVS Vent & Recirc and CR Charcoal/HEPA Filters Efficiencies

HEPA Filter:

In-place penetration testing acceptance criteria for the safety related HEPA filters are as follows:

FRVS Vent HEPA Filter – in-laboratory testing penetration < 0.05% (Ref. 10.6.1)

FRVS Recirc HEPA Filter – in-laboratory testing penetration < 0.05% (Ref. 10.6.10)

CREF HEPA Filter – in-laboratory testing penetration < 0.05% (Ref. 10.6.13)

GL 99-02 (Ref 10.3) requires a safety factor of at least 2 should be used to determine the filter efficiencies to be credited in the design basis accident.

Testing penetration (%) = (100% - η)/safety factor = (100% - η)/2

Where η = HEPA filter efficiency to be credited in the analysis

0.05% = (100% - η)/2

0.1% = (100% - η)

η = 100% - 0.1% = 99.9%

Conservatively, the HEPA filter efficiency of 99% is credited in the analysis

Charcoal Filter

In-place penetration testing acceptance criteria for the safety related Charcoal filters are as follows:

FRVS Vent Charcoal Filter – in-laboratory testing methyl iodide penetration < 2.5% (Ref. 10.6.2)

FRVS Recirc Charcoal Filter – in- laboratory testing methyl iodide penetration < 10% (Ref. 10.6.11)

CREF Recirc Charcoal Filter – in- laboratory testing methyl iodide penetration < 0.5% (Ref. 10.6.14)

Testing methyl iodide penetration (%) = $(100\% - \eta)$/safety factor = $(100\% - \eta)/2$

Where η = charcoal filter efficiency to be credited in the analysis

FRVS Recirc Charcoal Filter

$10\% = (100\% - \eta)/2$

$20\% = (100\% - \eta)$

$\eta = 100\% - 20\% = 80\%$

FRVS Vent Charcoal Filter

$2.5\% = (100\% - \eta)/2$

$5\% = (100\% - \eta)$

$\eta = 100\% - 5\% = 95\%$

Since the FRVS recirc and FRVS vent are in series (Ref. 41), the combined charcoal filter efficiency would be:

$\eta = [1 - (1 - 0.80)(1 - 0.95)] \times 100\% = [1 - (0.2 \times 0.05)] \times 100\% = [1 - 0.01] \times 100\% = 99\%$

CR Charcoal Filter

$0.5\% = (100\% - \eta)/2$

$1\% = (100\% - \eta)$

$\eta = 100\% - 1\% = 99\%$

Safety Grade Filter	Filter Efficiency Credited (%)		
	Aerosol	Elemental	Organic
FRVS Vent	99	99	99
FRVS Recirc	99	80	80
Control Room	99	99	99

Table 16
HCGS RADTRAD Nuclide Inventory File

Isotope	Core Activity (Ci) A	Core Thernal Power (MWt) B	Core Activity (Ci/MWt) A/B	Isotope	Core Activity (Ci) A	Core Thernal Power (MWt) B	Core Activity (Ci/MWt) A/B
CO-58*	5.287E+05	3458	.1529E+03	TE-131M	1.383E+07	3458	.3999E+04
CO-60*	6.328E+05	3458	.1830E+03	TE-132	1.333E+08	3458	.3855E+05
KR-85	1.157E+06	3458	.3346E+03	I-131	9.406E+07	3458	.2720E+05
KR-85M	2.788E+07	3458	.8063E+04	I-132	1.356E+08	3458	.3920E+05
KR-87	5.454E+07	3458	.1577E+05	I-133	1.917E+08	3458	.5543E+05
KR-88	7.691E+07	3458	.2224E+05	I-134	2.122E+08	3458	.6136E+05
RB-86	1.743E+05	3458	.5040E+02	I-135	1.792E+08	3458	.5181E+05
SR-89	9.386E+07	3458	.2714E+05	XE-133	1.869E+08	3458	.5406E+05
SR-90	9.213E+06	3458	.2664E+04	XE-135	5.420E+07	3458	.1567E+05
SR-91	1.274E+08	3458	.3684E+05	CS-134	1.869E+07	3458	.5406E+04
SR-92	1.352E+08	3458	.3910E+05	CS-136	5.122E+06	3458	.1481E+04
Y-90	9.555E+06	3458	.2763E+04	CS-137	1.363E+07	3458	.3943E+04
Y-91	1.184E+08	3458	.3423E+05	BA-139	1.745E+08	3458	.5046E+05
Y-92	1.357E+08	3458	.3924E+05	BA-140	1.677E+08	3458	.4850E+05
Y-93	1.533E+08	3458	.4433E+05	LA-140	1.728E+08	3458	.4998E+05
ZR-95	1.566E+08	3458	.4528E+05	LA-141	1.594E+08	3458	.4611E+05
ZR-97	1.599E+08	3458	.4623E+05	LA-142	1.554E+08	3458	.4493E+05
NB-95	1.561E+08	3458	.4515E+05	CE-141	1.573E+08	3458	.4549E+05
MO-99	1.739E+08	3458	.5028E+05	CE-143	1.513E+08	3458	.4374E+05
TC-99M	1.522E+08	3458	.4402E+05	CE-144	1.178E+08	3458	.3408E+05
RU-103	1.503E+08	3458	.4345E+05	PR-143	1.456E+08	3458	.4210E+05
RU-105	1.071E+08	3458	.3098E+05	ND-147	6.294E+07	3458	.1820E+05
RU-106	6.074E+07	3458	.1757E+05	NP-239	2.050E+09	3458	.5928E+06
RH-105	9.970E+07	3458	.2883E+05	PU-238	3.658E+05	3458	.1058E+03
SB-127	1.034E+07	3458	.2989E+04	PU-239	3.890E+04	3458	.1125E+02
SB-129	3.080E+07	3458	.8908E+04	PU-240	4.995E+04	3458	.1444E+02
TE-127	1.024E+07	3458	.2961E+04	PU-241	1.774E+07	3458	.5129E+04
TE-127M	1.359E+06	3458	.3930E+03	AM-241	2.455E+04	3458	.7099E+01
TE-129	3.030E+07	3458	.8764E+04	CM-242	7.032E+06	3458	.2034E+04
TE-129M	4.517E+06	3458	.1306E+04	CM-244	5.764E+05	3458	.1667E+03

* CO-58 & CO-60 activities are obtained from RADTRAD User's Manual, Table 1.4.3.2-3 (Ref. 10.2)

Table 17

Elemental Iodine Deposition Velocity - MSIV Leakage

Time	Temp Degree* F	Temp Degree K	(2809/T) -12.5	Deposition Velocity cm/sec	Deposition Velocity m/sec
0	340	444.26	-6.18	0.002076	2.076E-05
3	320	433.15	-6.01	0.002442	2.442E-05
6	250	394.26	-5.38	0.004630	4.630E-05
24	208	370.93	-4.93	0.007248	7.248E-05
96	180	355.37	-4.60	0.010096	1.010E-04
240	170	349.82	-4.47	0.011446	1.145E-04
480	150	338.71	-4.21	0.014896	1.490E-04
720					

* From Design Input 5.3.1.8, Table 6

Table 18

Elemental Iodine Deposition Rate - MSIV Failed Line

Time Hr	Deposition Velocity m/sec A	Main Steam Line Total Surface Area (m^2) B	Main Steam Line Total Volume (m^3) C	Elemental Iodine Removal Rate (hr^{-1}) (AxB)x3600/C
0	2.076E-05	247.77	39.58	0.4679
3	2.442E-05	247.77	39.58	0.5503
6	4.630E-05	247.77	39.58	1.0433
24	7.248E-05	247.77	39.58	1.6333
96	1.010E-04	247.77	39.58	2.2752
240	1.145E-04	247.77	39.58	2.5796
480	1.490E-04	247.77	39.58	3.3570
720				

A From Table 17
B & C From Section 7.2.5

Table 19

Elemental Iodine Deposition Rate - MSIV Intact Lines

Time	Deposition Velocity	Main Steam Line		Elemental Iodine Removal Rate
		Total Surface Area	Total Volume	
Hr	m/sec	(m^2)	(m^3)	(hr^{-1})
	A*	B	C	(AxB)x3600/C
0	2.076E-05	262.46	41.83	0.4690
3	2.442E-05	262.46	41.83	0.5516
6	4.630E-05	262.46	41.83	1.0457
24	7.248E-05	262.46	41.83	1.6371
96	1.010E-04	262.46	41.83	2.2805
240	1.145E-04	262.46	41.83	2.5855
480	1.490E-04	262.46	41.83	3.3647
720				

A From Table 17

B & C From Section 7.2.6

Table 20

Elemental Iodine Resuspension Rate - MSIV Leakage

Post-LOCA Time (hr)	Temp Degree F	Temp Degree K	-600/T	Resuspension Rate (hr^{-1})
0	340	444.26	-1.35	0.0403
3	320	433.15	-1.39	0.0389
6	250	394.26	-1.52	0.0340
24	208	370.93	-1.62	0.0309
96	180	355.37	-1.69	0.0287
240	170	349.82	-1.72	0.0280
480	150	338.71	-1.77	0.0265
720				

Resuspension Rate $(sec)^{-1}$ $= 2.32\ (2.00) \times 10^{-5}\ e^{-600/T} = 4.32 \times 10^{-5}\ e^{-600/T}$

Resuspension Rate $(hr)^{-1}$ $= 4.32 \times 3600 \times 10^{-5}\ e^{-600/T}$

Table 21

Net Elemental Iodine Removal Rate - MSIV Failed Line

Post-LOCA Time (hr)	Temp Degree F	Iodine Removal Rate A (hr-1)	Iodine Resuspension Rate B (hr-1)	Net Iodine Removal Rate $\lambda_f = A - B$ (hr-1)
0	340	0.4679	0.0403	0.4276
3	320	0.5503	0.0389	0.5114
6	250	1.0433	0.0340	1.0094
24	208	1.6333	0.0309	1.6024
96	180	2.2752	0.0287	2.2465
240	170	2.5796	0.0280	2.5516
480	150	3.3570	0.0265	3.3305
720				

A From Table 18
B From Table 20

Table 22

Net Elemental Iodine Removal Rate - Intact Lines

Post-LOCA Time (hr)	Temp Degree F	Iodine Removal Rate A (hr-1)	Iodine Resuspension Rate B (hr-1)	Net Iodine Removal Rate $\lambda_i = A - B$ (hr-1)
0	340	0.4690	0.0403	0.4287
3	320	0.5516	0.0389	0.5127
6	250	1.0457	0.0340	1.0118
24	208	1.6371	0.0309	1.6062
96	180	2.2805	0.0287	2.2518
240	170	2.5855	0.0280	2.5575
480	150	3.3647	0.0265	3.3383
720				

A From Table 19
B From Table 20

Table 23

Elemental Iodine Removal Efficiency - MSIV Failed Line

Post-LOCA Time (hr)	**Temp Degree F**	**Net Iodine Removal Rate λ_f (hr-1)**	**Elemental Iodine Removal Efficiency B (%)**
0	340	0.4276	34.79
3	320	0.5114	40.03
6	250	1.0094	63.56
24	208	1.6024	79.86
96	180	2.2465	89.42
240	170	2.5516	92.20
480	150	3.3305	96.42
720			

λ_f From Table 21
$B = 1-e^{-\lambda_f}$

Table 24

Elemental Iodine Removal Efficiency - Intact Lines

Post-LOCA Time (hr)	Temp Degree F	Net Iodine Removal Rate λ_i (hr-1)	Elemental Iodine Removal Efficiency B (%)
0	340	0.4287	34.87
3	320	0.5127	40.11
6	250	1.0118	63.64
24	208	1.6062	79.94
96	180	2.2518	89.48
240	170	2.5575	92.25
480	150	3.3383	96.45
720			

A From Table 22
$B = 1\text{-}e^{-\lambda_i}$

Table 25

Post-LOCA Containment Leakage Activity in CR With Charcoal/HEPA Filters (Ci)									
Isotope	**0-0.33**	**0.33-0.5**	**0.5-2**	**2-4**	**4-8**	**8-24**	**24-96**	**96-720**	**Total**
Co-58	0.00E+00	0.00E+00	2.94E-08	7.97E-09	3.00E-10	0.00E+00	0.00E+00	0.00E+00	3.76E-08
Co-60	0.00E+00	0.00E+00	3.52E-08	9.55E-09	3.61E-10	5.75E-14	0.00E+00	0.00E+00	4.51E-08
Kr-85	3.31E-05	4.77E-05	1.21E-02	2.31E-02	4.82E-02	3.67E-02	1.43E-02	9.79E-03	1.44E-01
Kr-85m	1.14E-03	1.61E-03	3.23E-01	4.52E-01	5.08E-01	3.25E-02	1.84E-07	0.00E+00	1.32E+00
Kr-87	1.83E-03	2.40E-03	2.69E-01	1.73E-01	4.07E-02	5.05E-06	0.00E+00	0.00E+00	4.87E-01
Kr-88	2.73E-03	3.77E-03	6.63E-01	7.77E-01	6.10E-01	9.35E-03	0.00E+00	0.00E+00	2.07E+00
Rb-86	1.48E-06	1.40E-06	3.08E-07	6.99E-08	2.60E-09	0.00E+00	0.00E+00	0.00E+00	3.26E-06
Sr-89	0.00E+00	0.00E+00	4.26E-05	1.16E-05	4.35E-07	6.88E-11	5.94E-12	2.69E-12	5.46E-05
Sr-90	0.00E+00	0.00E+00	3.02E-06	8.20E-07	3.09E-08	4.94E-12	4.44E-13	2.87E-13	3.87E-06
Sr-91	0.00E+00	0.00E+00	4.79E-05	1.12E-05	3.17E-07	0.00E+00	0.00E+00	0.00E+00	5.95E-05
Sr-92	0.00E+00	0.00E+00	3.47E-05	5.65E-06	7.67E-08	0.00E+00	0.00E+00	0.00E+00	4.04E-05
Y-90	0.00E+00	0.00E+00	3.16E-08	8.41E-09	3.04E-10	0.00E+00	0.00E+00	0.00E+00	4.03E-08
Y-91	0.00E+00	0.00E+00	5.20E-07	1.41E-07	5.32E-09	8.42E-13	0.00E+00	0.00E+00	6.67E-07
Y-92	0.00E+00	0.00E+00	3.93E-07	7.21E-08	1.24E-09	0.00E+00	0.00E+00	0.00E+00	4.66E-07
Y-93	0.00E+00	0.00E+00	5.76E-07	1.36E-07	3.91E-09	0.00E+00	0.00E+00	0.00E+00	7.17E-07
Zr-95	0.00E+00	0.00E+00	6.85E-07	1.86E-07	7.00E-09	1.11E-12	0.00E+00	0.00E+00	8.78E-07
Zr-97	0.00E+00	0.00E+00	6.50E-07	1.63E-07	5.21E-09	0.00E+00	0.00E+00	0.00E+00	8.18E-07
Nb-95	0.00E+00	0.00E+00	6.47E-07	1.76E-07	6.60E-09	1.04E-12	0.00E+00	0.00E+00	8.29E-07
Mo-99	0.00E+00	0.00E+00	9.15E-06	2.43E-06	8.81E-08	1.19E-11	0.00E+00	0.00E+00	1.17E-05
Tc-99m	0.00E+00	0.00E+00	6.41E-06	1.38E-06	3.29E-08	0.00E+00	0.00E+00	0.00E+00	7.82E-06
Ru-103	0.00E+00	0.00E+00	7.09E-06	1.92E-06	7.23E-08	1.14E-11	9.72E-13	0.00E+00	9.08E-06
Ru-105	0.00E+00	0.00E+00	3.46E-06	6.87E-07	1.39E-08	0.00E+00	0.00E+00	0.00E+00	4.16E-06
Ru-106	0.00E+00	0.00E+00	1.93E-06	5.23E-07	1.97E-08	3.14E-12	2.81E-13	1.74E-13	2.47E-06
Rh-105	0.00E+00	0.00E+00	3.39E-06	8.86E-07	3.09E-08	0.00E+00	0.00E+00	0.00E+00	4.31E-06
Sb-127	0.00E+00	0.00E+00	8.80E-06	2.36E-06	8.63E-08	1.22E-11	1.22E-11	0.00E+00	1.12E-05
Sb-129	0.00E+00	0.00E+00	2.25E-05	4.43E-06	8.81E-08	0.00E+00	0.00E+00	0.00E+00	2.70E-05
Te-127	0.00E+00	0.00E+00	7.46E-06	1.75E-06	4.90E-08	0.00E+00	0.00E+00	0.00E+00	9.25E-06
Te-127m	0.00E+00	0.00E+00	1.16E-06	3.16E-07	1.19E-08	1.89E-12	2.10E-13	0.00E+00	1.49E-06
Te-129	0.00E+00	0.00E+00	8.81E-06	7.24E-07	2.50E-09	0.00E+00	0.00E+00	0.00E+00	9.54E-06
Te-129m	0.00E+00	0.00E+00	7.64E-06	2.07E-06	7.79E-08	1.23E-11	1.04E-12	0.00E+00	9.79E-06

Table 25 (Cont'd)

Post-LOCA Containment Leakage Activity in CR With Charcoal/HEPA Filters

(Ci)

Isotope	0-0.333	0.33-0.5	0.5-2	2-4	4-8	8-24	24-96	96-720	Total
Te-131m	0.00E+00	0.00E+00	1.40E-05	3.64E-06	1.25E-07	0.00E+00	0.00E+00	0.00E+00	1.78E-05
Te-132	0.00E+00	0.00E+00	1.41E-04	3.76E-05	1.37E-06	1.90E-10	9.02E-12	0.00E+00	1.80E-04
I-131	2.72E-03	2.58E-03	6.72E-04	1.75E-04	3.15E-05	9.83E-06	2.69E-06	1.97E-07	6.19E-03
I-132	3.62E-03	3.26E-03	5.45E-04	7.83E-05	4.27E-06	1.14E-08	0.00E+00	0.00E+00	7.51E-03
I-133	5.65E-03	5.33E-03	1.33E-03	3.27E-04	5.21E-05	1.01E-05	3.25E-07	0.00E+00	1.27E-02
I-134	4.81E-03	3.99E-03	3.20E-04	1.73E-05	1.33E-07	0.00E+00	0.00E+00	0.00E+00	9.14E-03
I-135	5.20E-03	4.84E-03	1.08E-03	2.31E-04	2.77E-05	1.71E-06	3.18E-10	0.00E+00	1.14E-02
Xe-133	7.16E-03	1.03E-02	2.59E+00	4.89E+00	9.98E+00	6.96E+00	1.82E+00	4.04E-02	2.63E+01
Xe-135	1.66E-03	2.36E-03	5.35E-01	8.76E-01	1.35E+00	3.03E-01	4.86E-04	0.00E+00	3.06E+00
Cs-134	4.46E-04	4.23E-04	9.33E-05	2.12E-05	7.92E-07	1.26E-10	1.13E-11	7.16E-12	9.84E-04
Cs-136	1.20E-04	1.13E-04	2.49E-05	5.64E-06	2.09E-07	3.21E-11	2.47E-12	0.00E+00	2.64E-04
Cs-137	2.67E-04	2.53E-04	5.58E-05	1.27E-05	4.74E-07	7.56E-11	6.80E-12	4.40E-12	5.89E-04
Ba-139	0.00E+00	0.00E+00	2.81E-05	2.79E-06	1.41E-08	0.00E+00	0.00E+00	0.00E+00	3.09E-05
Ba-140	0.00E+00	0.00E+00	7.54E-05	2.04E-05	7.63E-07	1.17E-10	8.96E-12	1.78E-12	9.66E-05
La-140	0.00E+00	0.00E+00	7.47E-07	1.96E-07	6.91E-09	0.00E+00	0.00E+00	0.00E+00	9.50E-07
La-141	0.00E+00	0.00E+00	5.02E-07	9.58E-08	1.79E-09	0.00E+00	0.00E+00	0.00E+00	5.99E-07
La-142	0.00E+00	0.00E+00	2.79E-07	3.09E-08	1.75E-08	0.00E+00	0.00E+00	0.00E+00	3.28E-07
Ce-141	0.00E+00	0.00E+00	1.72E-06	4.65E-07	1.75E-08	2.75E-12	1.62E-13	0.00E+00	2.20E-06
Ce-143	0.00E+00	0.00E+00	1.61E-06	4.18E-07	1.45E-08	0.00E+00	0.00E+00	0.00E+00	2.04E-06
Ce-144	0.00E+00	0.00E+00	1.12E-06	3.03E-07	1.14E-08	1.82E-12	2.04E-13	9.88E-14	1.43E-06
Pr-143	0.00E+00	0.00E+00	6.53E-07	1.77E-07	6.61E-09	0.00E+00	0.00E+00	0.00E+00	8.36E-07
Nd-147	0.00E+00	0.00E+00	2.91E-07	7.87E-08	2.94E-09	0.00E+00	0.00E+00	0.00E+00	3.73E-07
Np-239	0.00E+00	0.00E+00	2.13E-05	5.64E-06	2.03E-07	2.66E-11	0.00E+00	0.00E+00	2.71E-05
Pu-238	0.00E+00	0.00E+00	1.52E-09	4.12E-10	1.56E-11	2.48E-15	0.00E+00	0.00E+00	1.95E-09
Pu-239	0.00E+00	0.00E+00	3.85E-10	1.05E-10	3.95E-12	6.29E-16	5.66E-17	3.67E-17	4.93E-10
Pu-240	0.00E+00	0.00E+00	4.82E-10	1.31E-10	4.94E-12	7.88E-16	7.08E-17	4.59E-17	6.18E-10
Pu-241	0.00E+00	0.00E+00	8.29E-08	2.25E-08	8.50E-10	1.36E-13	1.22E-14	7.87E-15	1.06E-07
Am-241	0.00E+00	0.00E+00	3.37E-11	9.16E-12	3.46E-13	0.00E+00	0.00E+00	0.00E+00	4.32E-11
Cm-242	0.00E+00	0.00E+00	8.90E-09	2.42E-09	9.12E-11	0.00E+00	0.00E+00	0.00E+00	1.14E-08
Cm-244	0.00E+00	0.00E+00	4.81E-10	1.31E-10	4.93E-12	0.00E+00	0.00E+00	0.00E+00	6.16E-10

From RADTRAD Computer Run HAST1000CL02

Table 26

Post-LOCA Containment Leakage Activity in CR Without Charcoal/HEPA Filters (Ci)									
Isotope	0-0.333	0.33-0.5	0.5-2	2-4	4-8	8-24	24-96	96-720	Total
Co-58	0.00E+00	0.00E+00	9.30E-08	4.69E-08	2.18E-09	0.00E+00	0.00E+00	0.00E+00	1.42E-07
Co-60	0.00E+00	0.00E+00	1.11E-07	5.62E-08	2.61E-09	3.37E-13	2.34E-14	1.50E-14	1.70E-07
Kr-85	3.31E-05	4.77E-05	1.21E-02	2.31E-02	4.82E-02	3.67E-02	1.43E-02	9.79E-03	1.44E-01
Kr-85m	1.14E-03	1.61E-03	3.23E-01	4.52E-01	5.08E-01	3.25E-02	1.84E-07	0.00E+00	1.32E+00
Kr-87	1.83E-03	2.40E-03	2.69E-01	1.73E-01	4.07E-02	5.05E-06	0.00E+00	0.00E+00	4.87E-01
Kr-88	2.73E-03	3.77E-03	6.63E-01	7.77E-01	6.10E-01	9.35E-03	0.00E+00	0.00E+00	2.07E+00
Rb-86	1.48E-06	1.40E-06	1.17E-06	4.34E-07	1.89E-08	2.36E-12	0.00E+00	0.00E+00	4.50E-06
Sr-89	0.00E+00	0.00E+00	1.35E-04	6.80E-05	3.15E-06	4.03E-10	2.69E-11	1.22E-11	2.06E-04
Sr-90	0.00E+00	0.00E+00	9.56E-06	4.82E-06	2.24E-07	2.89E-11	2.01E-12	1.30E-12	1.46E-05
Sr-91	0.00E+00	0.00E+00	1.52E-04	6.61E-05	2.30E-06	9.21E-11	0.00E+00	0.00E+00	2.20E-04
Sr-92	0.00E+00	0.00E+00	1.10E-04	3.32E-05	5.56E-07	0.00E+00	0.00E+00	0.00E+00	1.44E-04
Y-90	0.00E+00	0.00E+00	1.00E-07	4.94E-08	2.20E-09	0.00E+00	0.00E+00	0.00E+00	1.52E-07
Y-91	0.00E+00	0.00E+00	1.65E-06	8.30E-07	3.85E-08	4.93E-12	0.00E+00	0.00E+00	2.52E-06
Y-92	0.00E+00	0.00E+00	1.24E-06	4.24E-07	9.01E-09	0.00E+00	0.00E+00	0.00E+00	1.68E-06
Y-93	0.00E+00	0.00E+00	1.82E-06	8.02E-07	2.83E-08	0.00E+00	0.00E+00	0.00E+00	2.65E-06
Zr-95	0.00E+00	0.00E+00	2.17E-06	1.09E-06	5.07E-08	6.49E-12	4.37E-13	0.00E+00	3.31E-06
Zr-97	0.00E+00	0.00E+00	2.06E-06	9.56E-07	3.77E-08	0.00E+00	0.00E+00	0.00E+00	3.05E-06
Nb-95	0.00E+00	0.00E+00	2.05E-06	1.03E-06	4.78E-08	6.09E-12	0.00E+00	0.00E+00	3.13E-06
Mo-99	0.00E+00	0.00E+00	2.90E-05	1.43E-05	6.38E-07	6.95E-11	0.00E+00	0.00E+00	4.39E-05
Tc-99m	0.00E+00	0.00E+00	2.03E-05	8.12E-06	2.38E-07	0.00E+00	0.00E+00	0.00E+00	2.86E-05
Ru-103	0.00E+00	0.00E+00	2.24E-05	1.13E-05	5.24E-07	6.67E-11	4.40E-12	1.80E-12	3.42E-05
Ru-105	0.00E+00	0.00E+00	1.09E-05	4.04E-06	1.01E-07	0.00E+00	0.00E+00	0.00E+00	1.51E-05
Ru-106	0.00E+00	0.00E+00	6.10E-06	3.07E-06	1.43E-07	1.84E-11	1.27E-12	7.86E-13	9.31E-06
Rh-105	0.00E+00	0.00E+00	1.07E-05	5.21E-06	2.24E-07	2.11E-11	0.00E+00	0.00E+00	1.62E-05
Sb-127	0.00E+00	0.00E+00	2.79E-05	1.38E-05	6.25E-07	7.14E-11	0.00E+00	0.00E+00	4.23E-05
Sb-129	0.00E+00	0.00E+00	7.12E-05	2.61E-05	6.38E-07	0.00E+00	0.00E+00	0.00E+00	9.79E-05
Te-127	0.00E+00	0.00E+00	2.36E-05	1.03E-05	3.55E-07	0.00E+00	0.00E+00	0.00E+00	3.42E-05
Te-127m	0.00E+00	0.00E+00	3.68E-06	1.86E-06	8.63E-08	1.11E-11	7.56E-13	4.15E-13	5.63E-06
Te-129	0.00E+00	0.00E+00	2.79E-05	4.26E-06	1.81E-08	0.00E+00	0.00E+00	0.00E+00	3.22E-05
Te-129m	0.00E+00	0.00E+00	2.42E-05	1.22E-05	5.64E-07	7.17E-11	4.69E-12	1.78E-12	3.69E-05

Table 30

Post-LOCA ESF Leakage Activity in CR With Charcoal/HEPA Filters (Ci)									
Isotope	**0-0.33**	**0.33-0.5**	**0.5-2**	**2-4**	**4-8**	**8-24**	**24-96**	**96-720**	**Total**
I-131	1.78E-03	1.69E-03	5.43E-04	3.76E-04	3.41E-04	1.28E-04	7.01E-05	5.04E-06	4.93E-03
I-132	2.37E-03	2.13E-03	4.40E-04	1.68E-04	4.62E-05	1.49E-07	0.00E+00	0.00E+00	5.16E-03
I-133	3.70E-03	3.49E-03	1.07E-03	7.01E-04	5.63E-04	1.32E-04	8.47E-06	0.00E+00	9.66E-03
I-134	3.15E-03	2.61E-03	2.58E-04	3.71E-05	1.44E-06	0.00E+00	0.00E+00	0.00E+00	6.06E-03
I-135	3.40E-03	3.17E-03	8.76E-04	4.95E-04	2.99E-04	2.23E-05	8.30E-09	0.00E+00	8.26E-03

From RADTRAD Run HAST1000ESF02

Table 31

Post-LOCA ESF Leakage Activity in CR Without Charcoal/HEPA Filters (Ci)										Total Activity C/HEPA Fltr (Ci)
Isotope	**0-0.33**	**0.33-0.5**	**0.5-2**	**2-4**	**4-8**	**8-24**	**24-96**	**96-720**	**Total**	
I-131	1.78E-03	1.69E-03	1.95E-03	1.67E-03	1.54E-03	5.82E-04	3.17E-04	2.28E-05	9.55E-03	4.62E-03
I-132	2.37E-03	2.13E-03	1.58E-03	7.45E-04	2.09E-04	6.73E-07	0.00E+00	0.00E+00	7.03E-03	1.88E-03
I-133	3.70E-03	3.49E-03	3.85E-03	3.11E-03	2.55E-03	5.98E-04	3.84E-05	0.00E+00	1.73E-02	7.67E-03
I-134	3.15E-03	2.61E-03	9.27E-04	1.65E-04	6.53E-06	0.00E+00	0.00E+00	0.00E+00	6.86E-03	8.01E-04
I-135	3.40E-03	3.17E-03	3.14E-03	2.20E-03	1.36E-03	1.01E-04	3.76E-08	0.00E+00	1.34E-02	5.10E-03

From RADTRAD Run HAST1000ESF03

Table 32

Post-LOCA Total Iodine & Aerosol Activity in CR Charcoal/HEPA Filters (Ci)

Isotope	Containment Leakage 0-720	MSIV Leakage 0-720	ESF Leakage 0-720	Total Iodine & Aerosol
Co-58	1.04E-07	0.00E+00	0.00E+00	1.04E-07
Co-60	1.25E-07	0.00E+00	0.00E+00	1.25E-07
Kr-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	1.24E-06	0.00E+00	0.00E+00	1.24E-06
Sr-89	1.51E-04	0.00E+00	0.00E+00	1.51E-04
Sr-90	1.07E-05	0.00E+00	0.00E+00	1.07E-05
Sr-91	1.61E-04	0.00E+00	0.00E+00	1.61E-04
Sr-92	1.03E-04	0.00E+00	0.00E+00	1.03E-04
Y-90	1.11E-07	0.00E+00	0.00E+00	1.11E-07
Y-91	1.85E-06	0.00E+00	0.00E+00	1.85E-06
Y-92	1.21E-06	0.00E+00	0.00E+00	1.21E-06
Y-93	1.94E-06	0.00E+00	0.00E+00	1.94E-06
Zr-95	2.43E-06	0.00E+00	0.00E+00	2.43E-06
Zr-97	2.23E-06	0.00E+00	0.00E+00	2.23E-06
Nb-95	2.30E-06	0.00E+00	0.00E+00	2.30E-06
Mo-99	3.22E-05	0.00E+00	0.00E+00	3.22E-05
Tc-99m	2.08E-05	0.00E+00	0.00E+00	2.08E-05
Ru-103	2.52E-05	0.00E+00	0.00E+00	2.52E-05
Ru-105	1.09E-05	0.00E+00	0.00E+00	1.09E-05
Ru-106	6.85E-06	0.00E+00	0.00E+00	6.85E-06
Rh-105	1.19E-05	0.00E+00	0.00E+00	1.19E-05
Sb-127	3.11E-05	0.00E+00	0.00E+00	3.11E-05
Sb-129	7.09E-05	0.00E+00	0.00E+00	7.09E-05
Te-127	2.50E-05	0.00E+00	0.00E+00	2.50E-05
Te-127m	4.14E-06	0.00E+00	0.00E+00	4.14E-06
Te-129	2.26E-05	0.00E+00	0.00E+00	2.26E-05
Te-129m	2.71E-05	0.00E+00	0.00E+00	2.71E-05

Table 32 (Cont'd)

Post-LOCA Total Iodine & Aerosol Activity in CR Charcoal/HEPA Filters (Ci)				
Isotope	Containment Leakage 0-720	MSIV Leakage 0-720	ESF Leakage 0-720	Total Iodine & Aerosol
Te-131m	4.89E-05	0.00E+00	0.00E+00	4.89E-05
Te-132	4.98E-04	0.00E+00	0.00E+00	4.98E-04
I-131	2.83E-03	2.64E-02	4.62E-03	3.39E-02
I-132	1.86E-03	2.07E-05	1.88E-03	3.76E-03
I-133	5.41E-03	2.12E-02	7.67E-03	3.43E-02
I-134	9.44E-04	0.00E+00	8.01E-04	1.75E-03
I-135	4.15E-03	3.33E-03	5.10E-03	1.26E-02
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	3.75E-04	0.00E+00	0.00E+00	3.75E-04
Cs-136	1.00E-04	0.00E+00	0.00E+00	1.00E-04
Cs-137	2.24E-04	0.00E+00	0.00E+00	2.24E-04
Ba-139	7.45E-05	0.00E+00	0.00E+00	7.45E-05
Ba-140	2.68E-04	0.00E+00	0.00E+00	2.68E-04
La-140	2.62E-06	0.00E+00	0.00E+00	2.62E-06
La-141	1.56E-06	0.00E+00	0.00E+00	1.56E-06
La-142	7.40E-07	0.00E+00	0.00E+00	7.40E-07
Ce-141	6.10E-06	0.00E+00	0.00E+00	6.10E-06
Ce-143	5.61E-06	0.00E+00	0.00E+00	5.61E-06
Ce-144	3.96E-06	0.00E+00	0.00E+00	3.96E-06
Pr-143	2.32E-06	0.00E+00	0.00E+00	2.32E-06
Nd-147	1.03E-06	0.00E+00	0.00E+00	1.03E-06
Np-239	7.49E-05	0.00E+00	0.00E+00	7.49E-05
Pu-238	5.40E-09	0.00E+00	0.00E+00	5.40E-09
Pu-239	1.37E-09	0.00E+00	0.00E+00	1.37E-09
Pu-240	1.71E-09	0.00E+00	0.00E+00	1.71E-09
Pu-241	2.95E-07	0.00E+00	0.00E+00	2.95E-07
Am-241	1.20E-10	0.00E+00	0.00E+00	1.20E-10
Cm-242	3.16E-08	0.00E+00	0.00E+00	3.16E-08
Cm-244	1.71E-09	0.00E+00	0.00E+00	1.71E-09

8.0 RESULTS SUMMARY

The results of AST analyses are summarized in the following sections:

8.1 Licensing Basis Analysis

The results of analyses, which establish licensing basis for the deletion of MSIVSS, increased MSIV, and CR unfiltered inleakage, are summarized in the following table:

Post-LOCA Activity Release Path	Post-LOCA TEDE Dose (Rem)		
	Receptor Location		
	Control Room	EAB	LPZ
Containment Leakage	4.50E-01	3.41E-01 (3.2 hr)	1.10E-01
ESF Leakage	2.85E-01	3.51E-02 (0 hr)	1.19E-02
MSIV Leakage	3.48E+00	1.92E+00 (9.3 hr)	3.67E-01
Containment Purge	0.00E+00	0.00E+00	0.00E+00
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	0.00E+00	0.00E+00	0.00E+00
CR Filter Shine	2.46E-03*	0.00E+00	0.00E+00
Total	**4.22E+00**	**2.30E+00**	**4.89E-01**
Allowable TEDE Limit	**5.00E+00**	**2.50E+01**	**2.50E+01**
	RADTRAD Computer Run No.		
Containment Leakage	HAST900CL00	HAST900CL00	HAST900CL00
ESF Leakage	HAST900ESF00	HAST900ESF00	HAST900ESF00
MSIV Leakage	HAST900MS00	HAST900MS00	HAST900MS00

* CR filter shine dose due to the CR unfiltered inleakage of 1000 cfm with the RADTRAD default nuclide inventory file (NIF) will bound that due to the CR unfiltered inleakage of 900 cfm with the plant-specific NIF.

8.2 V&V of RADTRAD V3.02 Code

The comparison of results of RADTRAD3.02 and HABIT1.0 codes are shown in the following table:

Comparison of Control Room Doses - Licensing Basis Case					
Dose ID	Post-LOCA Control Room Dose (Rem)				Dose Variation (%)
	HABIT	RADTRAD			
	Cont+ESF+MSIV	Cont+MSIV	ESF	Total	
Thyroid	3.3634E-01	2.2593E-01	1.1570E-01	3.4163E-01	+1.57%
Whole Body	2.3027E-02	2.3392E-02	6.0348E-06	2.3398E-02	+1.61%
Comparison of Exclusion Area Boundary Doses - Licensing Basis Case					
Dose ID	Post-LOCA Exclusion Area Boundary Dose (Rem)				Dose Variation (%)
	HABIT	RADTRAD			
	Cont+ESF+MSIV	Cont+MSIV	ESF	Total	
Thyroid	1.2500E+02	9.1537E+01	3.3431E+01	1.2497E+02	-0.02%
Whole Body	1.3480E+00	1.2283E+00	1.4366E-01	1.3720E+00	+1.78%
Comparison of Low Population Zone Doses - Licensing Basis Case					
Dose ID	Post-LOCA Low Population Zone Dose (Rem)				Dose Variation (%)
	HABIT	RADTRAD			
	Cont+ESF+MSIV	Cont+MSIV	ESF	Total	
Thyroid	1.6820E+01	1.1796E+01	5.0355E+00	1.6832E+01	+0.07%
Whole Body	2.4120E-01	2.3056E-01	1.5273E-02	2.4583E-01	+1.92%

9.0 CONCLUSIONS/RECOMMENDATIONS

9.1 CONCLUSIONS:

The results of analyses in Section 8 above indicate that the main steam isolation valve sealing system (MSIVSS) can be safely eliminated along with the increased MSIV leakage of 250 scfh and control room unfiltered inleakage of 900 cfm using the AST and guidance in the Regulatory Guide 1.183. Adherence to guidance in the RG 1.183 and use of the specific values and limits contained in the technical specifications and as-built post-accident performance of safety grade ESF functions provide the assurance of sufficient safety margin, including a margin to account for analysis uncertainties in the proposed uses of an AST and the associated facility modifications and changes to procedures.

The verification & validation of RADTRAD3.02 computer code (Section 8.2) demonstrates that the RADTRAD3.02 code produces the identical results within ± 2% margin of error compared to HABIT1.0 code for the same source terms, release mechanisms, and dose conversion factors. RADTRAD has been developed and tested by NRC in accordance with the requirements of ANSI/ANS-10.4-1987 in Reference 10.2, in Section 3, "Quality Assurance." In addition to the use of these programming standards, various program elements were tested and examined to insure program quality and ability to produce accurate and consistent results with HABIT 1.1 code.

10.0 REFERENCES

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

10.2 S.L. Humphreys et al., "RADTRAD V3.02: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998.

10.3 USNRC, "Laboratory Testing of Nuclear-Grade Activated Charcoal," NRC Generic Letter 99-02, June 3, 1999.

10.4 10 CFR 50.67, "Alternate Source Term."

10.5 Calculation No. H-1-ZZ-MDC-1879, Rev 0, Control Room χ/Qs For FRVS Vent, RB Truck Bay, TB Louvers, and SPV Using ARCON96 Code.

10.6 HCGS Technical Specifications:

10.6.1 Specification 4.6.5.3.1.c.1, FRVS Vent HEPA Filter Testing Criterion

10.6.2 Specification 4.6.5.3.1.c.2, FRVS Vent Charcoal Filter Testing Criterion

10.6.3 Specification 4.6.5.3.1.c.3, FRVS Vent HEPA/Charcoal Filter Flow Rate Testing Criterion

10.6.4 Specification 6.8.4.f, Primary Containment Leak Rate Testing Program

10.6.5 Bases ¾.6.2, Depressurization Systems

10.6.6 Specification 5.2.1, Containment Configuration

10.6.7 Specification 5.2.3, Secondary Containment

10.6.8 Specification 4.6.5.1, Secondary Containment Integrity

10.6.9 Specification 1.35, Rated Thermal Power.

10.6.10 Specification 4.6.5.3.2.c.1, FRVS Recirc HEPA Filter Testing Criterion

10.6.11 Specification 4.6.5.3.2.c.2, FRVS Recirc Charcoal Filter Testing Criterion

10.6.12 Specification 4.6.5.3.2.c.3, FRVS Recirc HEPA/Charcoal Filter Flow Rate Testing Criterion

10.6.13 Specification 4.7.2.c.1, Control Room Emergency Filtration System Surveillance Requirements

10.6.14 Specification 4.7.2.c.2, Control Room Emergency Filtration System Surveillance Requirements

10.6.15 Specification 4.7.2.c.3, Control Room Emergency Filtration System Surveillance Requirements

10.6.16 Specification 4.7.2.e.3, Control Room Emergency Filtration System Surveillance Requirements

10.6.17 Specification 3.6.1.2.c, Primary Containment Leakage Limiting Condition For Operation

10.6.18 Specification 3.6.1.8, Drywell and Suppression Chamber Purge System

10.6.19 HCGS Technical Specification Table 3.6.3-1, Primary Containment Isolation Valves.

10.7 Federal Guidance Report 11, EPA-5201/1-88-020, Environmental Protection Agency.

10.8 Federal Guidance Report 12, EPA-402- R-93-081, Environmental Protection Agency.

10.9 MicroShield Computer Code, V&V Version 5.05, Grove Engineering.

10.10 Draft Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Facility Operating License No. NPF-57. Subject: Increase of Allowable Main Steam Isolation Valve (MSIV) Leak Rate and Deletion of MSIV Sealing System (TAC No. MA9978).

10.11 Drawing No. 1-P-AB-01, Rev 18, System Isometric / Turbine Building Main Steam Lead.

10.12 Fabrication Isometric Main Steam Lead – Turbine Building Unit #1 Drawings:

a. 1-P-AB-001, Rev 11
b. 1-P-AB-002, Rev 9
c. 1-P-AB-003, Rev 9
d. 1-P-AB-004, Rev 9
e. 1-P-AB-011, Rev 11

10.13 Piping Area Drawings:

a. P-1703-1, Rev 3, Reactor Building Area 17, Plan EL 100'-2".
b. P-1704-1, Rev 2, Reactor Building Area 17, Plan EL 112'-0".
c. P-1705-1, Rev 2, Reactor Building Area 17, Plan EL 121'-7-1/2".
d. P-1712-1, Rev 2, Reactor Building Area 17, Section B17 – B17.
e. P-1713-1, Rev 4, Reactor Building Area 17, Section C17 – C17.
f. P-1403-1, Rev 2, Reactor Building Area 14, Plan At EL 102'-0".
g. P-1414-1, Rev 1, Reactor Building Area 14, Section D14 – D14.

10.14 Piping Class Sheet Drawing No. 10855-P-0500::

10.14.a Sheet 16, Rev 9, Class DBB

10.14.b Sheet 17, Rev 7, Class DBC

10.14.c Sheet 24, Rev 7, Class DLA

10.15 GE-NE-T2300759-00-02, HCGS Containment Analysis With 100 ^{0}F SACS Temperature, September 1998.

10.16 Calculation No. 12-0025, Rev 3, "Drywell Volume & Torus Air & Water Volumes."

10.17 Specification 10855-M-786 (Q), Rev 11, Technical Specification For HVAC Air Filter Systems, Seismic Category I For The Hope Creek Generating Station.

10.18 Procedure HC.RA-AP.ZZ-0051(Q), Rev. 1, Leakage Reduction Program

10.19 Calculation No. GU-0013, Rev. 4, Filtration Recirculation and Ventilation System Exhaust Rate

10.20 Drawing M-76-1, Rev. 18, P&ID Reactor Building Air Flow Diagram

10.21 CR961030231 Act. 0010 Response, Secondary Containment

10.22 NRC Report AEB-98-03, " Assessment of Radiological Consequences For the Perry Pilot Plant Application Using the Revised (NUREG-1465) Source Term.

10.23 MSIV Leakage Iodine Transport Analysis By J.E. Cline & Associates, March 26, 1991, Contract NRC-03-87-029, Task Order 75

10.24 NUREG/CR-2713, Vapor Deposition Velocity Measurements and Correlations for I_2 and CsI, May 1982.

10.25 Calculation No. No. H-1-ZZ-MDC-0364, Rev 0, Drywell Temperature After Recirculation Line Break..

10.26 EQE International, Inc., Report No. 200235-R-01, November 12, 1998, Hope Creek Nuclear Plant Main Steam Isolation System Alternate Leakage Treatment Pathway Seismic Evaluation.

10.27 General Arrangement Drawings:

- 10.27.a P-0006-0, Rev 7, Plan EL 153'-0" and 162'-0"
- 10.27.b P-0011-0, Rev 5, Sections C-C & D-D

10.28 Equipment Location Drawings:

- 10.28.a P-0035-0, Rev 10, Service & Radwaste Area Plan EL 137'-0"
- 10.28.b P-0036-0, Rev 16, Service & Radwaste Area Plan EL 153'-0" & 155'-3"
- 10.28.c P-0055-0, Rev 15, Control & D/G Area, Plan EL 137'-0" & EL 146'-0" & EL 150'-0"
- 10.28.d P-0056-0, Rev 16, Control & D/G Area, Plan EL 155'-3" & EL 163'-6"

10.29 Auxiliary Bldg – Control Area Drawings:

- 10.29.a C-1317-0, Rev 22, Floor Plan EL 155'-3" Area 25
- 10.29.b C-1319-0, Rev 12, Floor Plan EL 155'-3" Area 26
- 10.29.c C-1321-0, Rev 5, Roof Plan EL 172-0" Area 25
- 10.29.d C-1323-0, Rev 4, Roof Plan EL 172-0" Area 26

10.30 Auxiliary Bldg – Control Area Drawings:

10.30.a C-1313-0, Rev 11, Floor Plan EL 137'-0" Area 25

10.30.b C-1315-0, SH 2, Rev 3, Floor Plan EL 137'-0" Area 26

10.31 Auxiliary Bldg – Diesel Generator Area Drawings:

10.31.a C-1413-0, Rev 20, Floor Plan EL 146'-0", EL 150'-0", EL 155'-3" Area 27

10.31.b C-1415-0, Rev 22, Floor Plan EL 146'-0", EL 150'-0", EL 155'-3" Area 28

10.32 Calculation No. H-1-ZZ-MDC-1820, Rev 0, Offsite Atmospheric Dispersion Factors.

10.33 Calculation No. H-1-ZZ-MDC-1882, Rev 0, Control Room Envelope Volume.

10.34 Draft NEI 99-03, Control Room Habitability Guidance, February 2001.

10.35 Drawing No. C-0738-0, Rev 6, Reactor Building Dome Reinforcement Plan Section & Details.

10.36 NRC Safety Evaluation for Amendment No. 30.

10.37 Specification No 10855-P-0501, Rev 34, Line Index For The Hope Creek Generating Station.

10.38 American Air Filter Drawing No. M786(Q)-5(1), Rev 10, Housing Assy Filter (Control Room Emergency Filter).

10.39 HVAC Area Drawings:

10.39.a P-9266-1, Rev 25, Aux Bldg Area 26, Plan At EL 155'-3"& 163'-6"

10.39.b P-9256-1, Rev 24, Aux Bldg Area 25, Plan At EL 155'-3"& 175'-0"

10.39.c P-9267-1, Sheet 1 of 4, Rev 17, Aux Building Area 25 & 26 Sections

10.40 U.S. NRC Standard Review Plan 6.4, Control Room Habitability System.

10.41 P&ID M-57-1, Rev 36, Containment Atmosphere Control.

10.42 P&ID M-78-1, Rev 9, Containment Hydrogen Recombination System.

10.43 Calculation No. H-1-ZZ-MDC-1866, Hope Creek Post-Accident pH.

10.44 Order No. 80028003, Confirmatory Inputs For H-1-ZZ-MDC-1880.

10.45 HCGS Core Inventory by Westinghouse, Hope Creek Calculation No. (Later)

11.0 Tracking of Items Required Confirmation

Some of critical design inputs are used in this analysis with bounding value assumptions, which required to be confirmed as soon as design input information is available. The following Orders will track the completion of required actions:

11.1 Order No.80028003

The isotopic core inventory used in this analysis is obtained from the RADTRAD nuclide inventory file and shown in the Design Input 5.3.1.3, Table 1 and Table 16 of this calculation. This isotopic core inventory should be compared with the plant-specific isotopic core inventory to determine the validity of Design Input 5.3.1.3.

11.2 Order No. 80028003

A pH value of 7.0 or greater is assumed for the suppression pool water inventory to take a credit of the chemical forms of radioiodine released to the containment to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide (Assumption 4.5). The pH of suppression pool water should be compared with the plant-specific pool water pH to determine the validity of pH assumption.

12.0 ATTACHMENTS

2 Diskettes with the following electronic files:

Calculation No: H-1-ZZ-MDC-1880, Rev OIR1.
Comment Resolution Form 2 – John F. Duffy
ATTACHMENT A: HCGSMHA_DEF.txt For Cont., ESF, & MSIV Leakages
ATTACHMENT B: Cont. Leakage RADTRAD Input File HAST900CL00.PSF
ATTACHMENT C: Cont. Leakage RADTRAD Output File HAST900CL00.o0
ATTACHMENT D: ESF Leakage RADTRAD Input File HAST900ESF00.PSF
ATTACHMENT E: ESF Leakage RADTRAD Output File HAST900ESF00.o0
ATTACHMENT F: MSIV Leakage RADTRAD Input File HAST900MS00.PSF
ATTACHMENT G: MSIV Leakage RADTRAD Output File HAST900MS00.o0
ATTACHMENT H: Cont. LKG Without CR Filter Input File HAST1000CL03.PSF
ATTACHMENT I: Cont. LKG Without CR Filter Output File HAST1000CL03.O0
ATTACHMENT J: ESF LKG Without CR Filter Input File HAST1000ESF03.PSF
ATTACHMENT K: ESF LKG Without CR Filter Output File HAST1000ESF03.O0
ATTACHMENT L: MSIV LKG Without CR Filter Input File HAST1000MS03.PSF
ATTACHMENT M: MSIV LKG Without CR Filter Output File HAST1000MS03.O0
ATTACHMENT N: CR Filter Shine Dose MicroShield Input/Output File
ATTACHMENT O: RADTRAD/HABIT1.0 V&V Files

99%oa.DSG	HCI30ESF00.O0
99%oacb.INP	HCI30ESF00.PSF
99%oacb.SPD	HCGSTID_DEF
99%oat5a.INP	HCCLI30MS00.PSF
99%oat5a.NUC	
99%oat5a.TAB	
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99%xxxcb.SPD	
99%xxx5a.INP	
99%xxxt5a.NUC	
99%xxxt5a.TAB	
HCCLI30MS00.O0	

2 Diskettes With Various Electronic Files

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

CALCULATION NO: H-1-ZZ-MDC-1879

Control Room χ/Qs for FRVS, RBTB, TBL, & SPV Using ARCON96 Code

CALC NO.: H-1-ZZ-MDC-1879 REVISION: 0IR0	CALCULATION COVER SHEET	Page 1 of 30
CALC. TITLE:	Control Room χ/Qs For FRVS, RBTB, TBL, & SPV Using ARCON96 Code	

# SHTS (CALC):	30	# ATT / # SHTS:	6	# IDV/50.59 SHTS:		# TOTAL SHTS:	

CHECK ONE:

☐ FINAL ☐ INTERIM (Proposed Plant Change) ☒ FINAL (Future Confirmation Req'd) ☐ VOID

SALEM OR HOPE CREEK: ☐ Q – LIST ☒ IMPORTANT TO SAFETY ☐ NON-SAFETY RELATED
HOPE CREEK ONLY: ☒Q ☐Qs ☐Qsh ☐F ☐R

☐ STATION PROCEDURES IMPACTED, IF SO CONTACT SYSTEM MANAGER
☐ CDs INCORPORATED (IF ANY):

DESCRIPTION OF CALCULATION REVISION (IF APPL.):

N/A

PURPOSE:

The purpose of this calculation is to determine 95% atmospheric dispersion factors (χ/Qs) at the Hope Creek Generating Station (HCGS) control room (CR) air intake due to the post-accidental releases from the Filtration Recirculation and Ventilation System (FRVS) Vent, Reactor Building Truck Bay (RBTB), Turbine Building Louver (TBL), and South Plant Vent (SPV) (Non-Loss of offsite power (LOOP) Event).

The control room χ/Qs are calculated using the NRC-Sponsored computer code ARCON96 and 7-year HCNGS plant specific meteorological data. All releases are assumed to be ground-level releases (zero exit velocity) with proper elevations of the release points to take the credit of appropriate site-specific meteorological data.

Additionally, ARCON96 computer code is verified by running the code test cases and validated by comparing their results. The 10 CFR 50.59 evaluation for DCP 4EC-3513, Page No. 1, Rev 1, applies to this documentation, which is CD P605.

CONCLUSIONS:

The 95% atmospheric dispersion factors χ/Qs for the FRVS, reactor building truck bay, turbine building louver, and south plant potential release points are summarized in the Sections 8.1 through 8.4. All releases are assumed to be ground-level releases with the appropriate elevations and vent release (mixed mode) is not credited. These χ/Qs should be used for the design basis accident analyses based on the potential release paths.

The verification & validation of ARCON96 computer code (Section 8.5) demonstrates that the ARCON96 code produces the consistent results for the test cases.

	Printed Name / Signature	Date
ORIGINATOR/COMPANY NAME:	Gopal J. Patel/NUCORE	04/12/01
PEER REVIEWER/COMPANY NAME:	R. Yewdall/PSEG Robert F. Yewdall	04/12/01
VERIFIER/COMPANY NAME:	J. Duffy/PSEG John Duffy	04/12/01
PSEG SUPERVISOR APPROVAL:	R. DeNight/PSEG B L Barkley for RWD	4-19-2001

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

REVISION HISTORY

Revision	Description
0	Original Issue

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1 through 67	0

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01					

1.0 PURPOSE

The purpose of this calculation is to determine 95% atmospheric dispersion factors (χ/Qs) (relative concentrations) at the Hope Creek Generating Station (HCGS) control room (CR) air intake due to the post-accidental releases from the following locations:

1. Filtration Recirculation and Ventilation System (FRVS) Vent Release
2. Reactor Building Truck Bay (RBTB)
3. Turbine Building Louver (TBL)
4. South Plant Vent (SPV) (Non-Loss of offsite power (LOOP) event)

The control room air intake χ/Qs are calculated using the NRC-sponsored computer code ARCON96 (Ref. 2) and 7-year HCNGS plant specific meteorological data (Ref. 1). The recommendation provided in the draft NEI 99-03 (Appendix D) for ARCON96 code to avoid use of the Vent Release Model (mixed mode release) in design basis accident applications is implemented. All releases are treated as ground-level releases (zero exit velocity) with the corresponding elevations of the release points to take the credit of appropriate site-specific wind speeds.

Additionally, ARCON96 computer code is verified by running the code test cases and validated by comparing the results.

2.0 BACKGROUND

The proposed elimination of the MSIV sealing system involves additional release paths that are not considered in our current LOCA radiological consequence analysis. The post-LOCA MSIV leakage activity can potentially be either released to environment via SPV when the offsite power is available or TBL during the LOOP.

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Similarly, the truck bay door is a release path during an FHA if a proposal to eliminate the requirement to maintain secondary containment integrity when irradiated fuel is being handled in the secondary containment and during core alterations and operations with a potential for draining the reactor vessel is implemented.

The atmospheric dispersion factors χ/Qs for these additional release paths are not readily available. Therefore, they are calculated in the following section using the ARCON96 computer code. Since the ARCON96 code is used for the new release paths, the existing FRVS χ/Qs are revised using the ARCON96 code to provide the consistent basis for the CR χ/Qs from all release locations.

3.0 ANALYTICAL APPROACH

The ARCON96 computer code (Ref. 10.2) was developed for the U.S. Nuclear Regulatory Commission Office Of Nuclear Reactor Regulation for use in control room habitability assessments. The ARCON96 code uses hourly meteorological data and recently developed methods for estimating dispersion in the vicinity of buildings to calculate relative concentrations at control room air intakes that would be exceeded no more than five percent of the time. These concentrations are calculated for averaging periods ranging from one hour to 30 days in duration.

The locations of release point of interest are configured with respect to the CR air intake location based on the dimensions given in the building arrangement drawings to establish the cross-section areas of structures, which control the downwind distance of building-wake (see Figures 1 through 4). Various receptor data (Ref. 10.2, pages 15 & 16) and source data (Ref. 10.2, pages 17 & 18) required for the ARCON96 input are established based on the plant-specific configuration. The meteorological data files were developed based on the 7-year site-specific meteorological measurements calculated for a period 1988 through 1994 (Ref. 10.1) and used for ARCON96 meteorological input (Ref. 10.2, pages 13 & 14). The required receptor and source input data are tabulated with Figures 1 through 4.

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4.0 DESIGN INPUT PARAMETERS

Parameter	Value	Reference
HCGS Meteorological Data	1988 – 1944 Measured Meteorological Data	Ref. 10.1 & 10.22
CR Air Intake Data	Figure 1	Ref. 10.5 thru 10.7
FRVS Source Data	Figure 1	Ref. 10.10 thru 10.16, & 10.19
RBTB Source Data	Figure 2	Ref 10.8 & 10.19
TBL Source Data	Figure 3	Ref. 10.8, 10.10, 10.19, & 10.21
SPV Source Data	Figure 4	Ref. 10.8 thru 10.15, & 10.19
Surface Roughness Length (meters)	0.2	Ref 10.4, Table D-1
Averaging Sector Width Constant	4.3	Ref 10.4, Table D-1
Lower Measurement Height For Met Data (ft)	33	Ref. 10.20
Upper Measurement Height For Met Data (ft)	150	Ref. 10.20
Unit of Wind Speed In Met Data	mph	Ref. 10.22

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5.0 ASSUMPTIONS

1. The FRVS vent release is assumed to be a ground-level point source release (equivalent to capped vent release with zero exit velocity) because the exhaust from the FRVS vent is disperse horizontally in downwind or upwind direction (Ref. 10.16) and it completely gets trapped in the building-wake.

2. The RBTB release is assumed to be a ground-level point source release (Ref. 10.8).

3. However, the release from the SPV is directly released to the environment with a high exit velocity (Ref. 10.7, 10.10, & 10.11), the SPV release is assumed to be ground-level point source release (equivalent to uncapped vent release with zero exit velocity) per guidance in the NEI 99-03 (Ref. 10.4, Appendix D).

4. The TBL release is assumed to be a ground-level release diffused source (equivalent to capped vent release with zero exit velocity) because the exhaust from the TBL is released through a large area (28' x 6') and dispersed horizontally in downwind direction (Ref. 10.13) and it completely gets trapped in the wake.

5. Minimum wind speed is assumed at 0.5 m/s. It is default value used in the ARCON96 code and is used for applying low wind speed correction for calm wind condition.

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6.0 METHODOLOGY

6.1 Release Through FRVS Vent

During a Loss of Coolant Accident (LOCA), the containment leakage and ESF leakage take place in the reactor building, which mix with the RB volume and release to the environment via FRVS vent (Ref. 10.17) (Figure 1).

6.1.1 Control Room Air Intake χ/Q – Release Through TB

The cross-section areas of the turbine building (for east to west wind direction) and reactor building (for south to north wind direction) will contribute to the building-wake diffusion, which control the distance downwind of the FRVS release point. The cross-section areas of the south exterior wall (below EL 132'-0") of reactor building, reactor building above EL 132'-0", and east exterior wall of turbine building below EL 132'-0"will contribute to the wake diffusion (Figure 1). The CR air intake is located at EL 155'-5" (Ref. 10.7c) in the west wall of auxiliary building (Refs 10.6). The ARCON96 receptor and source parameters are calculated in Section 7.1. The ARCON96 input and output for the FRVS vent χ/Qs are shown in Appendix A.

6.2 Release Through Reactor Building Truck Bay

The activity from the Fuel Handling Accident (FHA) directly releases to the environment when the truck bay door is opened during the refueling outage and FHA occurs in the reactor building (RB). The truck bay is located in the south wall of RB (Ref. 10.8) (see Figure 2).

6.2.1 Control Room Air Intake χ/Q Release Through RBTB

The release from the RB truck bay will be affected by the building wake of the reactor building. The cross-sectional area of reactor building will contribute to the building wake diffusion. The rectangle RB area below EL 132'-0' and cylindrical and dome of RB above EL 132'-0" will determine the building wake (Ref 10.8 &

10.16). The ARCON96 receptor and source parameters are calculated in Section 7.2. The ARCON96 input and output for the RBTB release χ/Qs are shown in Appendix B.

6.3 Release Through Turbine Building Lover (TBL)

The post-LOCA Main Steam Isolation Valve (MSIV) leakage travels through the main steam line and enters the turbine building (TB) and then releases to the environment. The most limiting situation is a LOCA with Loss Of Offsite Power (LOOP). In this case, the leakage from the MSIV will seep through the TB and release to the environment through the openings (louvers) in the TB. However, these louvers supply the fresh air and equipped with the dampers, which close when the supply fans are not operating to eliminate the unmonitored release to environment, they provide the most conservative release path with respect to the control room (CR) air intake (see Figure 3). The most limiting release location is the center louver located between columns 21 & 23 at Raw H above elevation 171'-0" (Ref. 10.21). The center TBL is 28' wide x 6' high (Ref. 10.21). The release takes place over a large area, therefore, it is considered a diffused source in the following analysis.

6.3.1 Control Room Air Intake χ/Q – Release Through TBL

The release from the TB louvers will be affected by the building wake of the turbine building, reactor building, and auxiliary building. The cross-sectional area of auxiliary building and reactor building will contribute to the building wake diffusion. The full width of reactor building and auxiliary building are considered to determine the building wake (Ref. 10.10). The ARCON96 receptor and source parameters are calculated in Section 7.3. The ARCON96 input and output for the FRVS vent χ/Qs are shown in Appendix C.

6.4 Release Through South Plant Vent (SPV)

The activity from the mechanical vacuum pumps (MVPs) discharge to the SPV during the Control Rod Drop Accident (Ref. 10.18, page 16). The SPV is located at Row H and center- line of reactor building (Refs 10.13 – 15 & 10.18) (see Figure 4) in vicinity of the TBL (see Figure 3).

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6.4.1 Control Room Air Intake χ/Q Release Through SPV

The SPV and TB release point are located in the same neighborhood with approximately same distance from the CR air intake (see Figures 3 & 4). Therefore, both release points will be affected by the same building wake. Contrary, the SPV located at EL 217'-0" (Refs 10.14 & 10.15), which is higher than TBL EL 171'-0" (Ref. 10.15). The activity from the SPV is directly released to environment with a high exit velocity, therefore, the exhaust from the SPV can rises above the building wake, which provide a better dilution and lower χ/Qs at the CR air intake. Reference 18 restricts the use vent release option of ARCON96 code in design basis accident application because the vent release model is appropriate for use in long term routine effluent calculations such as Offsite Dose Calculation Manual (ODCM) (Ref. 18, page D-6). Therefore, a ground-level point source release is used for the SPV with the proper elevation. The ARCON96 receptor and source parameters are calculated in Section 7.3. The ARCON96 input and output for the FRVS vent χ/Qs are shown in Appendix D.

6.5 V&V of ARCON96 Code

The test cases in Examples 1 through 4 and 5e are executed by DELL Computer Serial # FXQ9R, P.O. #B3-0951230 R33. The calculated results are compared with those in the ARCON96 User's Manual to demonstrate the consistency of results and ability of the code to produce the same results in the different operating environment and configuration (see Section 8.5).

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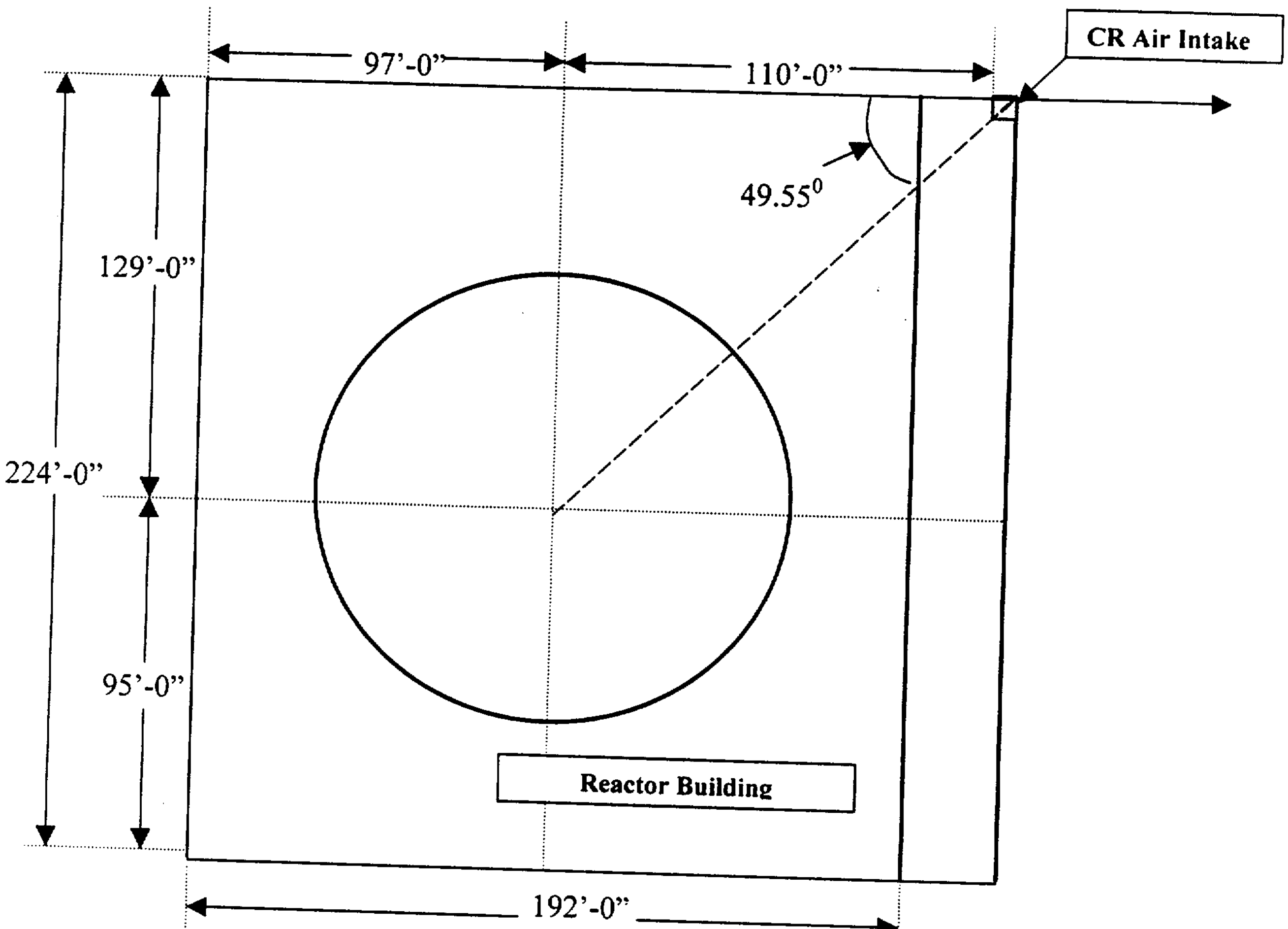

FIGURE 1: Relative Locations of FRVS Vent & CR Air Intake

Release Point	Distance To Receptor		Release Point Height		Dir To Source	Wake Area	CR Air Intake Ht
	ft	meter	ft	meter	Degree	m^2	meter
FRVS	169.53	51.69	198.67	60.6	130.45	3428.12	16.29

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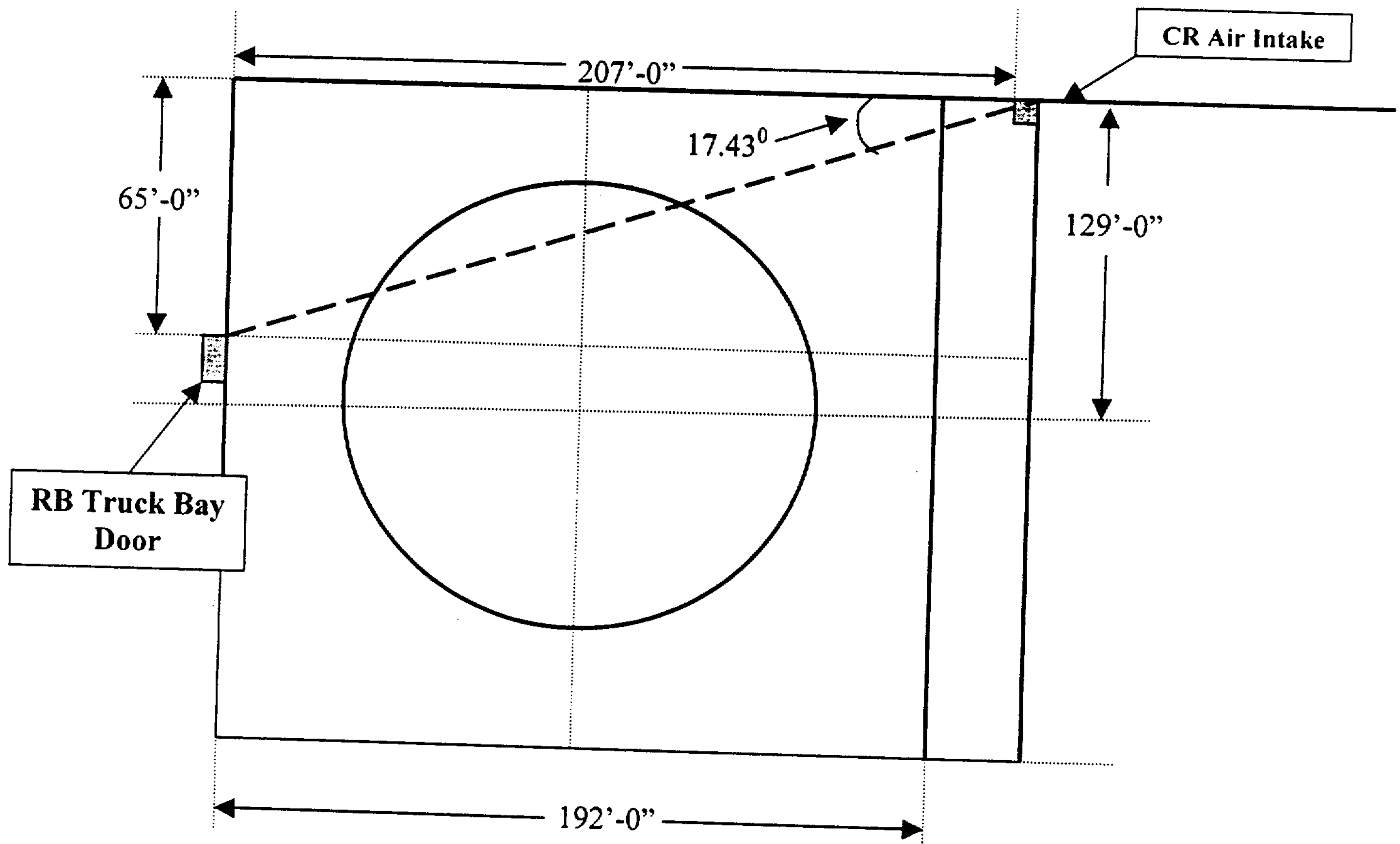

Figure 2: Relative Locations RB Truck Bay Door & CR Air Intake

Release Point	Distance To Receptor		Release Point Height		Dir To Source	Wake Area	CR Air Intake Ht
	ft	meter	ft	meter	Degree	m^2	meter
Truck Bay	216.96	66.15	0	0	162.57	3118.06	16.29

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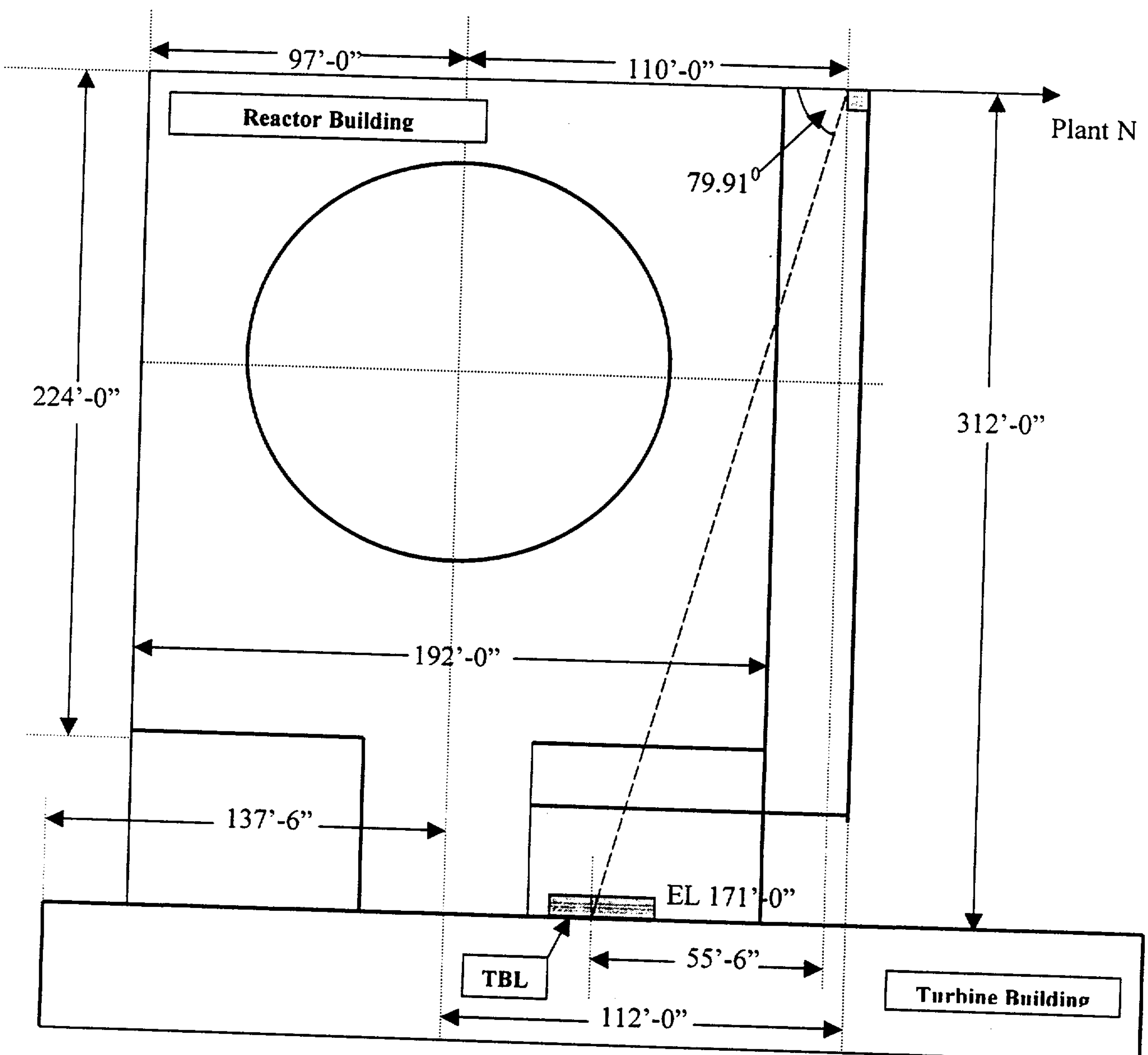

FIGURE 3: Relative Locations of TBL & CR Air Intake

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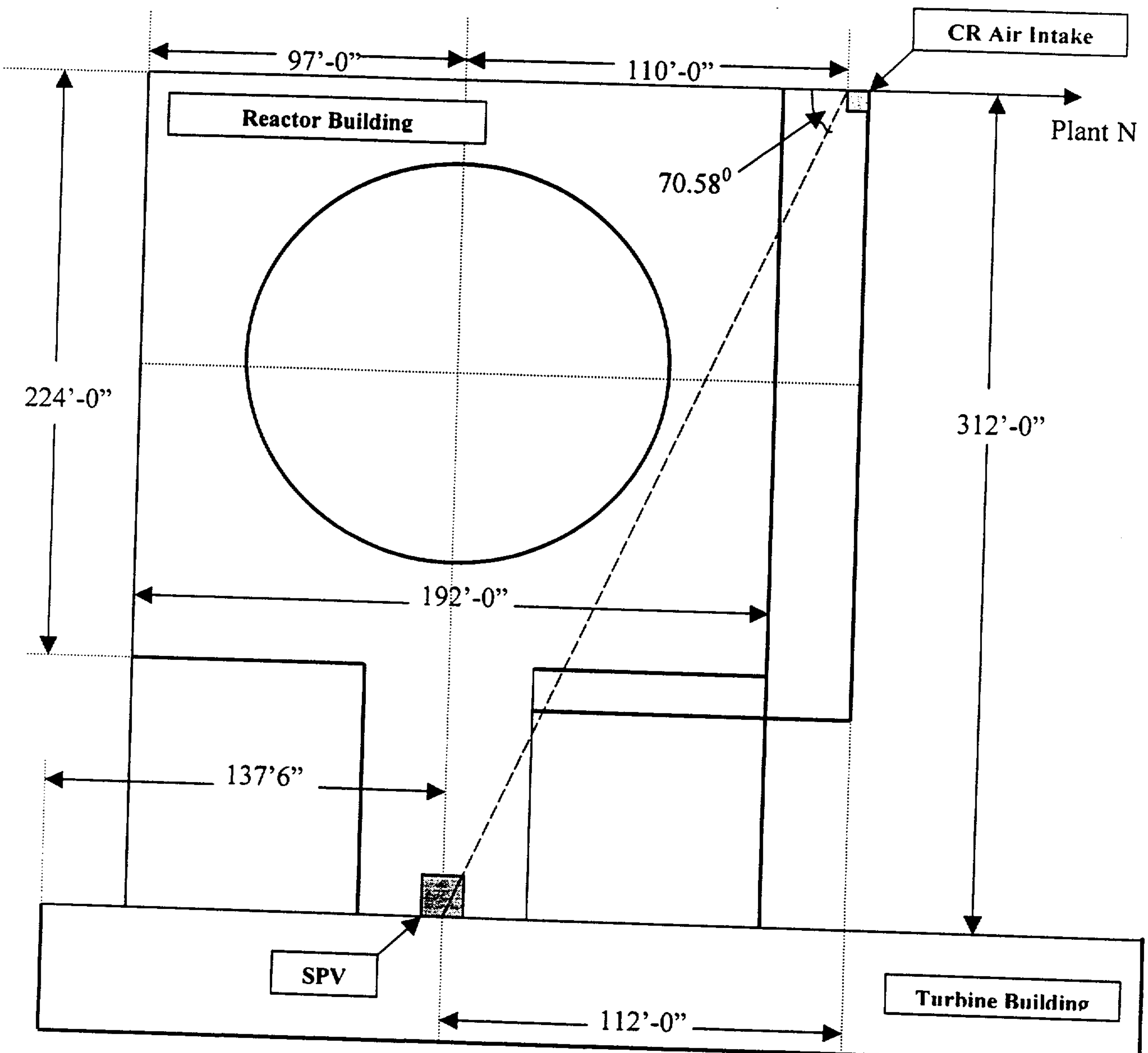

FIGURE 4: Relative Locations of SPV & CR Air Intake

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Release Point	Distance To Receptor		Release Point Height		Dir To Source	Wake Area	CR Air Intake Ht
	ft	meter	ft	meter	Degree	m^2	meter
TBL	316.9	96.62	69	21.03	100.09	4618.75	16.29
SPV	331.82	100.86	115	35.06	109.42	4618.75	16.29

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7.0 CALCULATIONS

The areas causing the wake diffusion are calculated in the following sections. The minor discrepancies exist in the various dimensions of the building. The conservative dimensions are used.

7.1 Wake Area For FRVS Release χ/Qs

The FRVS is located at the top of reactor building dome at elevation 300'-8" (Ref. 10.16).

For the wind direction from south to north, the south wall of reactor building will cause a wake diffusion for ground level wind (see Figure 1).

Reactor Building (RB) South Wall Surface Area

Width of Wall = 224'-0" (Ref. 10.10) (see Figure 1)

Top Elevation of Wall = 132'-0" (Ref. 10.14 & 10.15)

Horizontal Distance Between FRVS & CR Air Intake = 95'-0" (Ref. 10.10 & 11) + 17'-8" (Ref. 10.9) – 1'-8" (Ref. 10.7C). = 207'-0"

Grade Elevation of Wall = 102'-0" (Ref. 10.14 & 10.15)

Height of Wall = (132'-0" – 102'-0") = 30'-0"

Cross-Section Area of South Wall of RB = 224'-0" x 30'-0" = 6720 ft^2 = 624.63 m^2

For the wind direction from south to north, the reactor building cylindrical section and dome cross-section areas above elevation 132'-0" will cause a wake diffusion for the wind above RB.

Spring Line Elevation of Cylindrical Section = 250'-0" (Ref. 10.16)

Height of Cylindrical Section = 250'-0" – 132'-0" = 118'-0"

Diameter of Cylindrical Section = 2 x 85'-0" = 170'-0" (Ref. 10.16)

Cross-Section Area of Cylindrical Section = Diameter x Height

= 170'-0" x 118'-0" = 20060 ft^2 = 1864.59 m^2

FORM 2
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CALCULATION CONTINUATION SHEET

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ORIGINATOR, DATE	REV:	G. Patel, 04/10/01	0			
REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

Top Elevation of Dome = 300-8" (Ref. 10.16)

Dome Height Above Cylindrical Section = 300'-8" – 250'-0" = 50'-8"

Cross-Section Area of Dome

= $\pi/2$ x Radius x Height = $\pi/2$ x 85'-0" x 50'-8" = 6765.34 ft^2 = 628.84 m^2

Total Cross-Section Area of Cylindrical Section and Dome

= $\boxed{1864.59 \text{ m}^2 + 628.84 \text{ m}^2 = 2493.43 \text{ m}^2}$

<u>For the wind direction from east to west, the east wall of turbine building will cause a wake for ground level wind (see Figure 3).</u>

Turbine Building (TB) East Wall Surface Area:

Width of Wall = 137'-6' + 112'-0" = 249'-6" (Ref. 10.10)

Height of Wall Below EL 132'-0" = (132-0" – 102'-0") = 30'-0"

Cross-Section Area of East Wall Below EL 132'-0" = 249'-6" x 30'-0" = 7485 ft^2 = 695.74 m^2

FRVS Direction With Respect to CR Intake

Tan θ = 129/110 = 1.173, Therefore $\theta = \text{Tan}^{-1}$ 1.173 = 49.55^0

Orientation of FRVS Release with Respect to CR Air Intake, Considering South Wind 180^0 and North Wind 360^0 (Ref. 10.2, page 16).

FRVS Orientation = 180^0 - 49.55^0 = 130.45^0

Total Wake Cross-Section Area Perpendicular to Wind Direction to CR Air Intake

= $\boxed{624.63 \text{ m}^2 \times \text{Cos } 49.55^0 + 695.74 \text{ m}^2 \times \text{Sin } 49.55^0 + 2493.43 \text{ m}^2 = 3428.12 \text{ m}^2}$

Straight Line Distance Between FRVS and CR Air Intake = $[(129)^2 + (110)^2]^{-1/2}$ = 169.59 ft = 51.69 m

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01					

Elevation of CR Air Intake = 155'-5" (Ref. 10.7c, Section C).

Height of CR Air Intake = 155'-5" – 102'-0" = 53'-5"ft = 16.29 m

Elevation of FRVS Vent = 300'-8" (Ref. 10.16).

Height of FRVS Vent = 300'-8" – 102'-0" = 198'-8"ft = 60.6 m

7.2 Wake Area For Reactor Building Truck Bay Release χ/Qs

The location of reactor building truck bay (RBTB) with respect to CR air intake is shown in Figure 2 (Ref. 10.8 & 10.19). The RBTB location with respect to CR air intake is such that the wind from the south to north will predominantly carry effluent from the RBTB to the CR intake. Therefore, the wake diffusion effect for the wind from the east to west is not considered in the RBTB χ/Qs. Only the cross-sectional area perpendicular to wind from south to north is considered for the wake diffusion.

Total Cross-Section Area Perpendicular to Wind From South to North (see Section 7.1)

= Half Area of South wall of RB

= 129'-0" x (132'-0" –102'-0") = 129' x 30' = 3870 ft^2 = 359.72 m^2

= $\boxed{359.72\ m^2}$

RBTB Direction With Respect to CR Intake

Tan θ = 65/207 = 0.314, Therefore θ = Tan^{-1} 0.314 = 17.43^0

Orientation of RBTB Release with Respect to CR Air Intake, Considering South Wind 180^0 and North Wind 360^0 (Ref. 10.2, page 16).

Orientation = 180^0 - 17.43^0 = 162.57^0

Distance of RBTB From West Edge of RB = 65'-0" (Ref. 10.8)

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01		

Straight Line Distance Between RBTB and CR Air Intake = $[(65)^2 + (207)^2]^{-1/2}$ = 216.96 ft = 66.15 m.

Height of CR Air Intake = 155'-5" – 102'-0" = 53'-5"ft = 16.29 m

Height of RBTB = Ground Release = 0 m

7.3 Wake Area For Turbine Building Lovers (TBL) Release χ/Qs

Three large louvers exist in the exterior wall at Row H to facilitate the normal air intake to TB (Ref. 10.13 & 10.21). The center TBL provides direct and shortest release path to the CR air intake therefore, the TBL χ/Qs are analyzed based on the center louver location (see Figure 4). The TBL is located in the side exterior wall at the row H at elevation 171'-0" (Ref. 10 13 & 10.21). The release from the TBL is downwind direction, which is categorized as a capped vent release (Ref. 20, page 29). The TBL is located in vicinity of the SPV (see Figures 3 & 4). Therefore, the TBL experience the same wake effect with only difference in the elevation.

For the wind direction from east to west, the east wall of turbine building will cause a wake for ground level wind (see Figure 3).

Turbine Bldg Roof Elevation = 197'-0" (Ref 10.15)

Height of Turbine Bldg = 197'-0" – 102'-0" = 95'-0"

Auxiliary Bldg Width = 165'-0" (Ref. 10.10) = 50.30 m

RB Width Below Elevation 132'-0" = 192'-0" (Ref. 10.8)

Cross-Section Area of TB & Auxiliary Bldg = (RB Width + Auxiliary Bldg Width) x Height

= (192'-0" + 165'-0") x 95'-0" = 33915 ft^2 = 3152.42 m^2

Elevation of Cylindrical Section = 250'-0" (Ref. 10.16)

Height of Cylindrical Section Above TB Elevation = 250'-0" – 197'-0" = 53'-0"

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

Diameter of Cylindrical Section = 2 x 85'-0" = 170'-0" (Ref. 10.16)

Cross-Section Area of Cylindrical Section Above TB Elevation

= Diameter x Height = 170'-0" x 53'-0" = 9010 ft^2 = 837.49 m^2

Cross-Section Area of Dome = 628.84 m^2 (see Section 7.1)

Total Cross-Section Area Perpendicular to Wind From East to West

= $\boxed{3152.42\ m^2 + 837.49\ m^2 + 628.84\ m^2 = 4618.75\ m^2}$

Distance of Center of TBL From RB Center Line

= 6" + 18'-0" + 18'-0" + 4'-0" + 14'-0" = 54'-6"

Horizontal Distance Between TBL and CR Intake

= 110'-0" – 54'-6" = 55'-6"

TBL Direction With Respect to CR Intake

Tan θ = 312/55.5 = 5.621, Therefore θ = Tan^{-1} 5.621 = 79.91^0

Orientation of TBL Release with Respect to CR Air Intake, Considering South Wind 180^0 and North Wind 360^0 (Ref. 10.2, page 16).

TBL Orientation = 180^0 - 79.91^0 = 100.09^0

Straight Line Distance Between TBL and CR Air Intake = $[(312)^2 + (55.5)^2]^{-1/2}$ = 316.9 ft = 96.62 m

Elevation of TBL = 171'-0" (Ref. 10.14 & 10.15).

Elevation of TBL Above Grade = 171'-0" – 102'-0" = 69'-0" = 21.04 m

Effective Height of TBL = 6' (Ref. 10.21) = 6 ft / 3.28 ft/m = 1.829 m

Effective Width of TBL = 28' (Ref. 10.21) = 28 ft / 3.28 ft/m = 8.537 m

Vertical Diffusion Coefficient σ_z = Effective Height / 2 (Ref. 10.2, page 39) = 1.829 m /2 = 0.915 m

Horizontal Diffusion Coefficient σ_y = Effective Width / 4.3 (Ref. 10.2, page 39)

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= 8.537 m / 4.3 = 1.985 m.

7.4 Wake Area For South Plant Vent (SPV) Release χ/Qs

The SPV is located very close to the Row H at the centerline of RB and at elevation 217'-0" (Ref. 10. 7, 10.9, &10.10) (Figure 4). The SPV location with respect to CR air intake is such that the wind from the east to west will predominantly carry effluent from the SPV to the CR intake. Therefore, the wake diffusion effect from the east to west wind is considered in the SPV χ/Qs. Only the cross-sectional area perpendicular to wind direction is considered for the wake diffusion. The SPV is located in vicinity of the TBL (see Figures 3 & 4). Therefore, the SPV experience the same wake effect with only difference in the release elevation.

For the wind direction from east to west, the east wall of turbine building will cause a wake for ground level wind (see Figure 4).

Total Cross-Section Area Perpendicular to Wind From East to West (Section 7.3)

$$= \boxed{3152.42\ m^2 + 837.49\ m^2 + 628.84\ m^2 = 4618.75\ m^2}$$

SPV Direction With Respect to CR Intake

Tan θ = 312/110 = 2.836, Therefore $\theta = \text{Tan}^{-1}\ 2.836 = 70.58^0$

Orientation of SPV Release with Respect to CR Air Intake, Considering South Wind 180^0 and North Wind 360^0 (Ref. 10.2, page 16).

SPV Orientation = $180^0 - 70.58^0 = 109.42^0$

Straight Line Distance Between SPV and CR Air Intake = $[(312)^2 + (110)^2]^{-1/2}$ = 330.82 ft = 100.86 m

Elevation of SPV = 217'-0" (Ref. 10.14 & 10.15).

Height of SPV = 217'-0" – 102'-0" = 115'-0" ft = 35.06 m

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

8.0 RESULTS SUMMARY

8.1 Control Room χ/Qs For FRVS Releases

95% χ/Qs for the control room air intake due to the FRVS releases are summarized in the following Table:

Control Room 95% X/Qs For FRVS Release

Time Interval (hr)	CR X/Q (s/m^3)
0-2	1.26E-03
2-8	8.25E-04
8-24	3.35E-04
24-96	2.39E-04
96-720	1.76E-04

8.2 Control Room χ/Qs For Reactor Building Truck Bay Release

95% χ/Qs for the control room air intake due to the reactor building truck bay release are summarized in the following Table:

Control Room 95% X/Qs For RBTB Release

Time Interval (hr)	CR X/Q (s/m^3)
0-2	1.40E-03
2-8	1.19E-03
8-24	4.86E-04
24-96	3.33E-04
96-720	2.70E-04

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

8.3 Control Room χ/Qs For Turbine Building Louver Release

95% χ/Qs for the control room air intake due to the TB release are summarized in the following Table:

Control Room 95% X/Qs For TBL Release

Time Interval (hr)	CR X/Q (s/m^3)
0-2	6.00E-04
2-8	3.93E-04
8-24	1.49E-04
24-96	1.00E-04
96-720	7.66E-05

8.4 Control Room χ/Qs For South Plant Vent Release

95% χ/Qs for the control room air intake due to the TB release are summarized in the following Table:

Control Room 95% X/Qs For SPV Release

Time Interval (hr)	CR X/Q (s/m^3)
0-2	5.81E-04
2-8	3.90E-04
8-24	1.46E-04
24-96	9.56E-05
96-720	7.38E-05

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

8.5 Comparison of Results - ARCON96 Test Cases Vs V&V Cases

The ARCON96 test case examples are re-executed after ARCON96 have been installed on the network computer and the results are compared in the following Table with those in the ARCON96 User's Manual to demonstrate the code ability to produce the consistent results. The ARCON96 V&V input/output files are included in Attachment 11.1:

Comparison of ARCON96 Test Cases Vs V&V Cases

Example No.	Case Analyzed	Time Interval (Hr) X/Q Values (s/m^3) 0-2	2-8	8-24	24-96	96-720	Reference
1	Test	1.43E-03	1.04E-03	5.46E-04	4.49E-04	3.75E-04	Ref. 10.2, p23
	V&V	1.43E-03	1.04E-03	5.46E-04	4.49E-04	3.75E-04	Ex1vv_96
2	Test	1.94E-03	1.71E-03	7.74E-04	5.37E-04	2.74E-04	Ref. 10.2, p28
	V&V	1.94E-03	1.71E-03	7.74E-04	5.37E-04	2.74E-04	Ex2vv_96
3	Test	1.04E-02	8.12E-03	4.00E-03	3.03E-03	1.82E-03	Ref. 10.2, p30
	V&V	1.04E-02	8.12E-03	4.00E-03	3.03E-03	1.82E-03	Ex3vv_96
4	Test	1.53E-05	1.61E-05	3.67E-06	3.71E-06	3.55E-06	Ref. 10.2, p31
	V&V	1.53E-05	1.61E-05	3.67E-06	3.71E-06	3.55E-06	Ex4vv_96
5e	Test	6.73E-04	4.43E-04	1.40E-04	1.60E-04	1.38E-04	Ref. 10.2, p38
	V&V	6.73E-04	4.43E-04	1.40E-04	1.60E-04	1.38E-04	Ex5ev_96

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01					

9.0 CONCLUSIONS/RECOMMENDATIONS

The walk down to confirm the as-built configuration of the plant indicates that the control room air intake location analyzed in this calculation provides the air for both the filtered air intake and unfiltered inleakage paths (Ref 10.23). The 95% atmospheric dispersion factors χ/Qs for the FRVS, reactor building truck bay, turbine building louver, and south plant vent are summarized in the Section 8.0. All releases are assumed to be ground-level releases with the appropriate elevations and vent release (mixed mode) is not credited. These χ/Qs should be used for the design basis accident analyses based on the potential release paths.

The verification & validation of ARCON96 computer code (Section 8.5) demonstrates that the ARCON96 code produces the consistent results for the test cases.

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REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01					

10.0 REFERENCES

1. Hope Creek Nuclear Generating Meteorological Date Files (Attached CD.

2. NUREG/CR-6331 PNNL-10521, Rev 1, Atmospheric Relative Concentration in Building Wakes, May 1997.

3. Standard Review Plan 6.4, Rev. 2, Control Room Habitability System.

4. Draft NEI 99-03, February 2001, Appendix D, Atmospheric Dispersion.

5. HCNGS HVAC Area Drawings

   a. P-9256-1, Rev 24, "Aux Bldg Area 25 Plan At EL 155'-3" & EL 175'-0"."

   b. P-9266-1, Rev 25, "Aux Bldg Area 26 Plan At EL 155'-3" & EL 175'-0"."

   c. P-9267-1, Rev 17, "Aux Bldg Area 25 & 26 Sections."

   d. P-9268-1, Rev 16, "Aux Bldg Area 25 & 26 Sections and Plan At EL 178'-0"."

6. HCNGS Concrete Drawings – Auxiliary Bldg/Diesel Generator Area:

   a. C-1413-0, Rev 20, "Auxiliary Bldg – Diesel Generator Area Floor Plan, El 146'-0", EL 150'-0" and EL 155'-3" Area 27."

   b. C-1415-0, Rev 22, "Auxiliary Bldg – Diesel Generator Area Floor Plan, El 146'-0", EL 150'-0" and EL 155'-3" Area 28."

7. HCNGS Concrete Drawings Auxiliary Bldg/Control Area:

   a. C-1317-0, Rev 22, "Auxiliary Bldg – Control Area Floor Plan, El 155'-3" Area 25."

   b. C-1319-0, Rev 12, "Auxiliary Bldg – Control Area Floor Plan, El 155'-3" Area 26."

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c. C-1334-0, Rev 8, "Auxiliary Bldg, Control and Diesel Area, Misc Conc Plan and Section Details."

8. Reactor Building Floor Plan Drawings:

   a. C-0535-1, Rev 13, Elevation 102'-0" Area 19

   b. C-0538-1, Rev 10, Elevation 102'-0" Area 22

   c. C-0539-1, Rev 8, Elevation 102'-0" Area 23

   d. C-0540-1, Rev 11, Elevation 102'-0" Area 22

9. C-1077-0, Rev 6, "Aux Bldg, Radwaste & Service Area Roof Plan, El 172,-0" Area 32."
10. Drawing P-0000-0, Rev 9, Plant Design Drawing and Tag index.
11. Drawing P-0003-0, Rev 9, General Arrangement Plan – EL 102'-0".
12. Drawing P-0006-0, Rev 7, General Arrangement Plan – EL 153'-0" and 162'-0".
13 Drawing P-0007-0, Rev 7, General Arrangement Plan – EL 171'-0" and 201'-0".
14. Drawing P-0010-0, Rev 6, General Arrangement Plan – Section A-A & B-B.
15. Drawing P-0011-0, Rev 5, General Arrangement Plan – Section C-C & D-D.
16. Drawing C-0738-0, Rev 6, Reactor Building Dome Reinforcement Plan Section & details.
17. Air Flow Diagram M-76-1, Rev 19, Reactor Building Air Flow Diagram.
18. Calc No.H-1-CG-MDC-1795, Control Rod Drop Accident – Analysis Reconstitution.
19. Drawing C-1407-0, Rev 25, Auxiliary Bldg, Diesel Generator Area Floor Plan EL 102'-0" Area 28.

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ORIGINATOR, DATE	**REV:**	G. Patel, 04/10/01	0				
REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01					

20. Salem Nuclear Generating Station Drawing No. 239850, Rev 12, Meteorological Tower .
21. HVAC Area/Section Drawings:
    a. P-9106-1, Rev 14, Turbine Bldg Area 10 Plan at EL 171'-0"
    b. P-9107-1, Rev 6, HVAC Drawing Section Turbine Bldg Area 10 EL 171'-0"
    c. P-9116-1, Rev 15, Turbine Bldg Area 11 Plan at EL 171'-0"
    d. P-9107-1, Rev 9, HVAC Drawing Section Turbine Bldg Area 11 EL 171'-0"
22 Memo From Robert Yewdall to Gopal Patel, Dated 04/09/2001, NRP-0-015, Subject: Design Input ARCON96 Meteorological Data, Calc H-1-ZZ-MDC-1879 (Attachment 11.2).
23. E-mail From Barkley, Barry L. to Patel, Gopal J., Sent: Friday, April 06, 2001 5:05 PM, Subject: Confirmation of Openings in CR Building Exterior Wall (Attachment 11.3).

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CALCULATION CONTINUATION SHEET

	CALCULATION CONTINUATION SHEET			SHEET 30 of 30		
CALC. NO.: H-1-ZZ-MDC-1879			REFERENCE:			
ORIGINATOR, DATE	REV:	G. Patel, 04/10/01	0			
REVIEWER/VERIFIER, DATE		R. Yewdall, 04/10/01				

11.0 ATTACHMENTS

11.1 Diskettes with the following electronic files:

HCCALC: H-1-ZZ-MDC-1879, Rev 0.
ARCON96 Input Files
ARCON96 Input/Output File – Control Room χ/Qs For FRVS Vent
ARCON96 Input/Output File – Control Room χ/Qs For Reactor Building Truck Bay
ARCON96 Input/Output File – Control Room χ/Qs For Turbine Building Louvers
ARCON96 Input/Output File - Control Room χ/Qs For South Plant Vent
ARCON96 Input/Output File – ARCON96 Code Test Example 1 V&V Case
ARCON96 Input/Output File – ARCON96 Code Test Example 2 V&V Case
ARCON96 Input/Output File – ARCON96 Code Test Example 3 V&V Case
ARCON96 Input/Output File – ARCON96 Code Test Example 4 V&V Case
ARCON96 Input/Output File – ARCON96 Code Test Example 5e V&V Case
Met Data Files – 1988 through 1994
Design Verification Comments

11.2 Attachment B - Memo NRP-01-015, April 9, 2001 From Robert Yewdall To Gopal Patel Subject: Design Inputs ARCON96 Meteorological Data.

11.3 Attachment C - E-Mail From Barkley, Barry L. Sent: Friday, April 06, 2001 5:05 PM To: Patel, Gopal J. Subject: Confirmation of Openings in CR Building Exterior Wall

Diskettes With Various Electronic Files

TO: Gopal Patel
Contract Engineer – Supporting PSEG Nuclear LLC

FROM: Robert F. Yewdall (*Signed Original*)
Radiological Protection Support

SUBJECT: Design Input ARCON96 Meteorological Data
Calc H-1-ZZ-MDC-1979

DATE: April 9, 2001
NRP-01-015

The purpose of this memorandum is to provide the Technical Basis Design Document information for the meteorological data used in the Hope Creek Control Room radiological design basis dispersion calculation. The Technical Basis document 2001-02 is attached.

If you have any questions please call me (x2469)

Technical Basis ID 2001-02	Title Design Input ARCON96 Met Data Rev 0	Page 1 of 2
Originator Robert Yewdall Date 4/9/01 Reviewer Lucius Clark Date 4/9/01 Approved Robert Gary Date 4/10/01	Reference: Calc H-1-ZZ-MDC-1879	Radiation Protection Correspondence ID NRP-01-15

Purpose

The purpose of this Technical Basis Document is to provide required documentation for the meteorological data used to perform design basis calculation for Hope Creek Control Room dose analysis. Meteorological data files identified below are used in the computer program ARCON96 to produce relative concentration calculations (i.e. dispersion factors -□/Qs). The dispersion factors will be used to calculate activity concentration at the Hope Creek control room emergency air intake (identified as the receptor) from various source release points.

Method

A seven year meteorological database was used to perform dispersion calculations for Salem Unit 1 and Unit 2 control room design basis accident dose assessment. That analysis is identified as vendor calculation number 321035. The seven year period consists of meteorological data from 1988 through 1994. The same database is used for the Hope Creek calculation H-1-ZZ-MDC-1879.

The seven year database consists of seven separate files as follows:

CONMET88.MET
CONMET89.MET
CONMET90.MET
CONMET91.MET
CONMET92.MET
CONMET93.MET
CONMET94.MET

The file format is consistent with specification contained in NUREG/CR-6331, Rev 1, *Atmospheric Relative Concentrations in Building Wakes*. The above files consist of hourly data in the following format: (1x, A5, 3x, I3, I2, 2x, I3, I4, Ix, I2, 2x, I3, I4). The order of information in the record is the site ID, Julian day, hour of the day, wind direction @ 33', wind speed @ 33', stability class, wind direction @ 150' wind speed @ 150'. Wind data at 150' elevation is closest to the FRVS release point which is ~ 200'. As indicated in the above file format, all records are in integer values (e.g. wind speeds are times 10).

Wind speeds in the above files are in mile per hour (mph). The wind direction is from a bearing of north (i.e., 0 degrees are winds from the north).

Instrument levels on the on-site meteorological tower are: 33', 150' and 300'. Documentation for tower instrumentation is contained on drawing 75602-12, rev 0 (PSPB 147007) and the Hope Creek UFSAR, Section 2.3. The stability class is determined by delta temperature measurement between 300' and 33' using the P-G stability index found in Reg. Guides 1.21 and 1.23.

All meteorological data is collected and validated by approved Q'd procedures (i.e., ND.RS-TI.MET-1203(Q) Meteorological Monitoring System Data Collection, and ND.RS-TI.MET-1204(Q) Meteorological Monitoring System Data Validation.

While ARCON96 will accept as many as 10 years of data, the seven year database is sufficient. Seven years is more than the three to five years recommended in the draft NEI 99-03 document. The seven year DB is also identical to that used for the Salem Control Room analysis as discussed above.

References:

NEI 99-03, Control Room Habitability Assessment Guidance
NUREG/CR 6331, Rev 1, Atmospheric Relative Concentrations in Building Wakes
ND.RS-TI.MET-1203(Q) Meteorological Monitoring System Data Collection
ND.RS-TI.MET-1204(Q) Meteorological Monitoring System Data Validation
VTD 321035, Accident values At The Salem Generating Station Control Room Fresh Air Intakes, Exclusion Area Boundary And Low Population Zone, MES, 4/12/96

From: Barkley, Barry L.
Sent: Friday, April 06, 2001 5:05 PM
To: Patel, Gopal J.
Cc: Cichello, John P.; DeNight, Robert W.; Duffy, John F.; Yewdall, Robert F.
Subject: Confirmation of Openings in CR Building Exterior Wall

This memo is to document the walk-down that John Cichello, you (Gopal Patel), and I (Barry Barkley) performed of the plant to confirm the subject matter as follows.

1. There is no opening in the south wall of the Aux Building (Control) which houses the control room. This south wall is the wall closest to the reactor building (& FRVS vent). All of the openings in the south wall of the Aux Building (Diesel/Control) are openings in the Aux Building (Diesel) which is a separate building (walls & doors) with respect to leakage into the control room.

2. The door that provides access to the roof above the Tech Support Area EL 155'-3" is reasonably airtight with weather stripping around the opening. There are 2 other doors before you reach the 137'EL hall that reaches the CREF Boundary door at Room 5502.

3. The door that provides access to the roof (EL 172'-0") of the Aux Building (Control) is a Security Door R6, which requires security permission to open. This door is reasonably airtight with weather stripping around the opening. There is 1 other door before you reach the CREF boundary door at Room 5512.

Based on the above walk down, we determined that the CREF air intake provides the air supply for both the CR filtered air intake and unfiltered inleakage for the Hope Creek Plant. The copy of this e-mail will be attached to Calculation H-1-ZZ-MDC-1879 to confirm the subject matter.

Table 26 (Cont'd)

Post-LOCA Containment Leakage Activity in CR Without Charcoal/HEPA Filters (Ci)									
Isotope	**0-0.33**	**0.33-0.5**	**0.5-2**	**2-4**	**4-8**	**8-24**	**24-96**	**96-720**	**Total**
Te-131m	0.00E+00	0.00E+00	4.44E-05	2.14E-05	9.07E-07	8.08E-11	0.00E+00	0.00E+00	6.67E-05
Te-132	0.00E+00	0.00E+00	4.47E-04	2.21E-04	9.93E-06	1.11E-09	4.08E-11	0.00E+00	6.78E-04
I-131	2.72E-03	2.58E-03	2.48E-03	1.03E-03	1.58E-04	4.45E-05	1.22E-05	8.93E-07	9.02E-03
I-132	3.62E-03	3.26E-03	2.01E-03	4.59E-04	2.14E-05	5.15E-08	0.00E+00	0.00E+00	9.37E-03
I-133	5.65E-03	5.33E-03	4.90E-03	1.92E-03	2.61E-04	4.58E-05	1.47E-06	0.00E+00	1.81E-02
I-134	4.81E-03	3.99E-03	1.18E-03	1.01E-04	6.67E-07	0.00E+00	0.00E+00	0.00E+00	1.01E-02
I-135	5.20E-03	4.84E-03	4.00E-03	1.36E-03	1.38E-04	7.74E-06	1.44E-09	0.00E+00	1.55E-02
Xe-133	7.16E-03	1.03E-02	2.59E+00	4.89E+00	9.98E+00	6.96E+00	1.82E+00	4.04E-02	2.63E+01
Xe-135	1.66E-03	2.36E-03	5.35E-01	8.76E-01	1.35E+00	3.03E-01	4.86E-04	0.00E+00	3.06E+00
Cs-134	4.46E-04	4.23E-04	3.53E-04	1.32E-04	5.77E-06	7.39E-10	5.13E-11	3.24E-11	1.36E-03
Cs-136	1.20E-04	1.13E-04	9.43E-05	3.50E-05	1.52E-06	1.88E-10	1.12E-11	1.83E-12	3.64E-04
Cs-137	2.67E-04	2.53E-04	2.11E-04	7.87E-05	3.45E-06	4.43E-10	3.08E-11	1.99E-11	8.13E-04
Ba-139	0.00E+00	0.00E+00	8.89E-05	1.64E-05	1.02E-07	0.00E+00	0.00E+00	0.00E+00	1.05E-04
Ba-140	0.00E+00	0.00E+00	2.39E-04	1.20E-04	5.52E-06	6.87E-10	4.06E-11	6.40E-12	3.64E-04
La-140	0.00E+00	0.00E+00	2.36E-06	1.15E-06	5.00E-08	0.00E+00	0.00E+00	0.00E+00	3.57E-06
La-141	0.00E+00	0.00E+00	1.59E-06	5.63E-07	1.29E-08	0.00E+00	0.00E+00	0.00E+00	2.16E-06
La-142	0.00E+00	0.00E+00	8.85E-07	1.82E-07	1.40E-09	0.00E+00	0.00E+00	0.00E+00	1.07E-06
Ce-141	0.00E+00	0.00E+00	5.43E-06	2.74E-06	1.27E-07	1.61E-11	1.05E-12	0.00E+00	8.30E-06
Ce-143	0.00E+00	0.00E+00	5.08E-06	2.46E-06	1.05E-07	0.00E+00	0.00E+00	0.00E+00	7.65E-06
Ce-144	0.00E+00	0.00E+00	3.53E-06	1.78E-06	8.27E-08	1.07E-11	7.36E-13	4.48E-13	5.39E-06
Pr-143	0.00E+00	0.00E+00	2.07E-06	1.04E-06	4.79E-08	5.96E-12	0.00E+00	0.00E+00	3.15E-06
Nd-147	0.00E+00	0.00E+00	9.23E-07	4.63E-07	2.13E-08	2.63E-12	0.00E+00	0.00E+00	1.41E-06
Np-239	0.00E+00	0.00E+00	6.74E-05	3.32E-05	1.47E-06	1.56E-10	0.00E+00	0.00E+00	1.02E-04
Pu-238	0.00E+00	0.00E+00	4.81E-09	2.42E-09	1.13E-10	1.45E-14	1.01E-15	0.00E+00	7.34E-09
Pu-239	0.00E+00	0.00E+00	1.22E-09	6.15E-10	2.86E-11	3.68E-15	2.56E-16	1.66E-16	1.86E-09
Pu-240	0.00E+00	0.00E+00	1.53E-09	7.69E-10	3.58E-11	4.61E-15	3.21E-16	2.08E-16	2.33E-09
Pu-241	0.00E+00	0.00E+00	2.63E-07	1.32E-07	6.16E-09	7.94E-13	5.52E-14	3.57E-14	4.01E-07
Am-241	0.00E+00	0.00E+00	1.07E-10	5.39E-11	2.50E-12	3.23E-16	0.00E+00	0.00E+00	1.63E-10
Cm-242	0.00E+00	0.00E+00	2.82E-08	1.42E-08	6.60E-10	0.00E+00	0.00E+00	0.00E+00	4.31E-08
Cm-244	0.00E+00	0.00E+00	1.52E-09	7.67E-10	3.57E-11	4.60E-15	0.00E+00	0.00E+00	2.32E-09

From RADTRAD Computer Run HAST1000CL03

Table 27

Containment Leakage Total Activity on CR Charcoal/HEPA Filter (Ci)			
Isotope	**0-720**	**Isotope**	**0-720**
Co-58	1.04E-07	Te-131m	4.89E-05
Co-60	1.25E-07	Te-132	4.98E-04
Kr-85	0.00E+00	I-131	2.83E-03
Kr-85m	0.00E+00	I-132	1.86E-03
Kr-87	0.00E+00	I-133	5.41E-03
Kr-88	0.00E+00	I-134	9.44E-04
Rb-86	1.24E-06	I-135	4.15E-03
Sr-89	1.51E-04	Xe-133	0.00E+00
Sr-90	1.07E-05	Xe-135	0.00E+00
Sr-91	1.61E-04	Cs-134	3.75E-04
Sr-92	1.03E-04	Cs-136	1.00E-04
Y-90	1.11E-07	Cs-137	2.24E-04
Y-91	1.85E-06	Ba-139	7.45E-05
Y-92	1.21E-06	Ba-140	2.68E-04
Y-93	1.94E-06	La-140	2.62E-06
Zr-95	2.43E-06	La-141	1.56E-06
Zr-97	2.23E-06	La-142	7.40E-07
Nb-95	2.30E-06	Ce-141	6.10E-06
Mo-99	3.22E-05	Ce-143	5.61E-06
Tc-99m	2.08E-05	Ce-144	3.96E-06
Ru-103	2.52E-05	Pr-143	2.32E-06
Ru-105	1.09E-05	Nd-147	1.03E-06
Ru-106	6.85E-06	Np-239	7.49E-05
Rh-105	1.19E-05	Pu-238	5.40E-09
Sb-127	3.11E-05	Pu-239	1.37E-09
Sb-129	7.09E-05	Pu-240	1.71E-09
Te-127	2.50E-05	Pu-241	2.95E-07
Te-127m	4.14E-06	Am-241	1.20E-10
Te-129	2.26E-05	Cm-242	3.16E-08
Te-129m	2.71E-05	Cm-244	1.71E-09

Table 28

Post-LOCA MSIV Leakage Activity in CR With Charcoal/HEPA Filters (Ci)							
Isotope	**0-24**	**24-29.52**	**29.52--96**	**96-240**	**240-480**	**480-720**	**Total**
Kr-85	4.06E-02	1.37E-02	2.24E-02	1.65E-02	1.49E-02	1.35E-02	1.22E-01
Kr-85m	3.60E-02	5.18E-03	2.89E-07	0.00E+00	0.00E+00	0.00E+00	4.12E-02
Kr-87	5.60E-06	9.33E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.69E-06
Kr-88	1.04E-02	9.10E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.13E-02
I-131	4.89E-03	9.50E-04	1.22E-03	3.15E-04	9.62E-05	2.26E-05	7.50E-03
I-132	5.66E-06	2.13E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.87E-06
I-133	5.03E-03	8.29E-04	1.48E-04	5.26E-07	1.28E-10	0.00E+00	6.01E-03
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	8.51E-04	9.45E-05	1.45E-07	0.00E+00	0.00E+00	0.00E+00	9.45E-04
Xe-133	7.71E+00	2.53E+00	2.86E+00	9.53E-01	2.31E-01	5.58E-02	1.43E+01
Xe-135	3.35E-01	7.43E-02	7.64E-04	9.57E-09	0.00E+00	0.00E+00	4.10E-01

From RADTRAD Computer Run HAST1000MS02

Table 29

Post-LOCA MSIV Leakage Activity in CR Without Charcoal/HEPA Filters (Ci)								Total Activity C/HEPA Fltr (Ci)
Isotope	**0-24**	**24-29.52**	**29.52--96**	**96-240**	**240-480**	**480-720**	**Total**	
Kr-85	4.06E-02	1.37E-02	2.24E-02	1.65E-02	1.49E-02	1.35E-02	1.22E-01	0.00E+00
Kr-85m	3.60E-02	5.18E-03	2.89E-07	0.00E+00	0.00E+00	0.00E+00	4.12E-02	0.00E+00
Kr-87	5.60E-06	9.33E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.69E-06	0.00E+00
Kr-88	1.04E-02	9.10E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.13E-02	0.00E+00
I-131	2.21E-02	4.31E-03	5.53E-03	1.43E-03	4.36E-04	1.02E-04	3.39E-02	2.64E-02
I-132	2.56E-05	9.64E-07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.66E-05	2.07E-05
I-133	2.28E-02	3.76E-03	6.68E-04	2.38E-06	5.79E-10	5.79E-10	2.72E-02	2.12E-02
I-134	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	3.85E-03	4.28E-04	6.55E-07	0.00E+00	0.00E+00	0.00E+00	4.28E-03	3.33E-03
Xe-133	7.71E+00	2.53E+00	2.86E+00	9.53E-01	2.31E-01	5.58E-02	1.43E+01	0.00E+00
Xe-135	3.35E-01	7.43E-02	7.64E-04	9.57E-09	0.00E+00	0.00E+00	4.10E-01	0.00E+00

From RADTRAD Computer Run HAST1000MS03