Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381-2000

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Watts Bar Nuclear Plant, Unit 2
NRC Docket No. 50-391

Subject: Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Chapter 15.5 Fuel Handling Accident (FHA) Dose Analysis

References:
1. TVA letter to NRC dated June 27, 2011, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Response to Request for Additional Information (RAI) Regarding Accident Dose Analysis Basis"
2. TVA letter to NRC dated August 5, 2011, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Chapter 15.5 Design Basis Dose Analysis"
3. TVA letter to NRC dated September 15, 2011, "Watts Bar Nuclear Plant (WBN) Unit 2 – Final Safety Analysis Report (FSAR) – Chapter 15.5 Design Basis Dose Analysis"

This letter provides revised FSAR Design Basis Accident (DBA) dose analysis results for the FHA.

Enclosure 1 provides a discussion of changes to the FHA analysis currently described in the FSAR and a revised FSAR Section 15.5.6 and associated tables updated to reflect the new results. The changes in the dose results for this DBA were the result of the following three items:

D030

1. Changes in assumptions on the closure time for Auxiliary Building and Main Control Room Dampers in the normal ventilation system.
2. Alternate source term being used as the basis for the dose calculations for the FHA in the Auxiliary Building and for the FHA in containment when the equipment hatch is open.
3. The incorporation of meteorology data for the 20 year period of 1991 to 2010 as opposed to the 1976 to 1993 data used for licensing Unit 1.

Enclosure 2 provides a complete draft of FSAR Section 15.5 red-line showing the recent revisions to the analyses as provided in this letter and References 2 and 3. Enclosure 3 provides a clean copy of the draft FSAR section.

The submittal of the results of this analysis was a TVA commitment to NRC described in Reference 1. This letter closes Commitment 2 of Reference 2 to provide the FHA results and the commitment in Reference 3 to provide the FHA and the proposed FSAR Section 15.5. There are no new regulatory commitments in this letter.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23rd day of September, 2011.

Respectfully,

David Stinson
Watts Bar Unit 2 Vice President

Enclosures:

1. WBN Unit 2 Revised FSAR Section 15.5 Fuel Handling Accident Dose Analysis Results

2. WBN Unit 2 – Revised FSAR Section 15.5. – Red-Lined

3. WBN Unit 2 – Revised FSAR Section 15.5

cc (Enclosures):

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Enclosure 1

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

The Fuel Handling Accident (FHA) was revised to account for a change in the meteorology data used and due to changes to Main Control Room and Auxiliary HVAC damper closure times. The meteorology was updated to use the 20 year period of 1991 to 2010. The analysis for a dropped fuel assembly inside containment when the containment air locks and equipment hatch are closed continues to use the methodology of Regulatory Guide (RG) 1.25. The purge system is in operation and credit is taken for the HEPA and charcoal beds prior to purge system isolation. The purge system will automatically isolate on high radiation from radiation monitors located in the purge system exhaust. No credit is taken for this isolation in the analysis and all activity is released to the environment within two hours as specified in RG-1.25.

Alternate source term (AST) described in RG-1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following changes are assumed in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with that required by RG-1.183.
- The decontamination factors were changed to be consistent with those required by RG-1.183.
- New onsite (control room) and offsite atmospheric dispersion factors (X/Q) are used.
- The time to isolate the control room is increased from 20.6 seconds to 40 seconds.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the Containment or the spent fuel pool to the environment by the containment purge filters or the Auxiliary Building Gas Treatment System (ABGTS) is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than the normal Auxiliary Building exhaust. In addition, any release from the shield building stack would go through the purge system HEPA and charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the Auxiliary Building also isolates, and ABGTS is actuated. The release point for ABGTS is the shield building stacks, and the releases are filtered through HEPA and charcoal assemblies. Thus, the AST analysis for the FHA in the Auxiliary Building that considers no filtration and no Auxiliary Building isolation is conservative and acceptable as the basis for the containment open evaluation.

The following pages of Enclosure 1 provide the revised FSAR Section 15.5.6 on the FHA. The section has been divided into three subsections. The first provides the assumptions for the RG-1.25 closed containment analysis. The second subsection, 15.5.6.2, provides the new assumptions for the AST. This is new information. Section 3 provides a summary of the results. There were only minor editorial changes in the first five paragraphs. The AST results summary starts at paragraph six. Table 15.5-20 was modified to reflect the closed containment analysis. A new Table 15.5-20.a was added for the AST analysis. The results are shown in Table 15.5-23, and the changes are marked.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers three cases. The first case is for a Fuel Handling Accident inside containment with the containment closed and the Reactor Building Purge System operating. This analysis is discussed in Section 15.5.6.1 and is based on Regulatory Guide 1.25[11] and NUREG 5009[24]. The second case is for an accident in the spent fuel pool area located in the Auxiliary Building. This case is discussed in Section 15.5.6.2 and is evaluated using the Alternate Source Term based on Regulatory Guide 1.183[18], "Alternate Source Terms." The third case considered is an open containment case for an accident inside containment where there is open communication between the containment and the Auxiliary Building. This evaluation is discussed in Section 15.5.6.2 and is based on Regulatory Guide 1.183.

15.5.6.1 Fuel Handling Accident Based on Regulatory Guide 1.25

The parameters used for this analysis are listed in Table 15.5-20.

The bases for the Regulatory Guide 1.25 evaluation are:

(1) In the Regulatory Guide 1.25 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.25 analysis damage is assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009 [24] as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.

(5) Noble gases released in the containment are released through the Shield Building vent to the environment.

(6) In the Regulatory Guide 1.25 analysis the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

(7) A filter efficiency of-90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters is used since no relative humidity control is provided.

(8) No credit is taken for natural decay after the activity has been released to the atmosphere.

(9) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.2 Fuel Handling Accident Based on Regulatory Guide 1.183

The analysis of a postulated fuel handling accident in the Auxiliary Building refueling Area is based on Regulatory Guide 1.183. i.e., Alternate Source Terms (AST). The parameters used for this analysis are listed in Table 15.5-20.a.

The bases for evaluation are:

(1) In the Regulatory Guide 1.183 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.183 analysis, damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) The Regulatory Guide 1.183 analysis assumes all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) In the Regulatory Guide 1.183 analysis, the iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) In the Regulatory Guide 1.183 analysis, the overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) In the Regulatory Guide 1.183 analysis, all iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(10) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(11) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.3 Fuel Handling Accident Results

The radiation dose results of the Regulatory Guide 1.25 with the containment closed fuel handling accident (FHA) are given in Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel-handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch-thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch-thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment.

The thyroid, gamma, and beta doses for FHAs for the closed containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated using the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel and the thyroid dose is below the limit of 30 rem.

The radiation dose results of the Regulatory Guide 1.183 fuel handling accident (FHA) are given in Table 15.5-23. Alternate source term (AST) described in RG 1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following assumptions are used in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with that required by RG 1.183.

- The decontamination factors were changed to be consistent with those required by RG. 1.183.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the Containment or the spent fuel pool to the environment by the Containment Purge filters or the ABGTS is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust. In addition, any release from the shield building stack would go through the purge system HEPA and Charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the auxiliary building also isolates and ABGTS is actuated. The release point for ABGTS is the shield building stacks and the releases are filtered through HEPA and Charcoal assemblies. Thus AST analysis for the FHA in the Auxiliary Building that considers no filtration is conservative and acceptable as the basis for the containment open evaluation.

The thyroid, gamma, and beta doses for FHAs in the Auxiliary and the open containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body and less than the 10 CFR 50.67 limit of 25 rem TEDE. These doses are calculated using the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem and the 10CFR 50.67 limit of 5 rem TEDE.

Enclosure 1

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

Table 15.5-20
Parameters Used In Fuel Handling Accident Analysis
Regulatory Guide 1.25 Analysis

Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released	
elemental iodine	99.75%
methyl iodine	0.25%
Filter efficiencies	RBPVS (2)
elemental iodine	90%
methyl iodine	30%
Amount of mixing of activity in Auxiliary Building	None

Meteorology See Table 15.5-14 and Table 15A-2

(1) 10% of the total radioactive iodine except for 12% of I-131 and 10% of total noble gases, except for 14% for Kr-85, 5% for Xe-133 and 2% for Xe-135 in the damaged rods at the time of the accident.

(2) Reactor Building Purge Ventilation System

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

Table 15.5-20.a
Parameters Used In Fuel Handling Accident Analysis
Regulatory Guide 1.183 Analysis

Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool	
elemental iodine methyl iodine	99.85%(AST) 0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology See Table 15.5-14 and Table15A-2	

(1) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Enclosure 1

WBN Unit 2 Revised FSAR Section 15.5
Dose Analysis

Table 15.5-23

Doses From A Fuel Handling Accident (FHA) (rem)

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Auxiliary Building (rem) or In Containment – Containment Open (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	~~3.994E-01~~ 4.29E-01	~~9.278E-02~~1.20E-01	~~4.935E-01~~5.86E-01
Beta	~~1.177E+00~~ 1.19E+00	~~2.734E-01~~3.33E-01	~~4.068E+00~~4.68E+00
Thyroid - ICRP-30	~~1.577E+00~~ 5.51E+01	~~3.663E-01~~1.54E+01	~~1.540E+00~~1.32E+01
TEDE	2.38E+00	6.66E-01	1.02E-00

Doses from Fuel Handling Accident Regulatory Guide 1.25 Analyses

FHA in Reactor Building, Containment Closed (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	~~4.102E-01~~4.31E-01	~~9.529E-02~~1.21E-01	~~2.677E-01~~2.72E-01
Beta	~~1.182E+00~~1.24E+00	~~2.746E-01~~3.48E-01	~~2.207E+00~~2.25E+00
Thyroid - ICRP-30	~~39.42E+00~~4.15E+01	~~9.158E+00~~1.16E+01	~~5.209E+00~~6.81E+00

Enclosure 2

WBN Unit 2 – Revised FSAR Section 15.5
Red-Lined

15.5 ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15.5.1 Environmental Consequences of a Postulated Loss of AC Power to the Plant Auxiliaries

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the reactor coolant system (RCS) to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a parameter. This analysis incorporates assumptions of a Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent, and a realistic source term. Parameters used in both the realistic and conservative analyses are listed in Table 15.5-1.

The realistic assumptions that determine the equilibrium concentrations of isotopes in the secondary system are as follows:

(1) Primary coolant activity is associated with 0.125% defective fuel and is given in Table 11.1-7.

(2) The iodine partition factor in the steam generators is:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.01$$

(3) No noble gas is dissolved or contained in the steam generator water, i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off gas system.

(4) The 0-2 and 2-8 hour atmospheric dilution factors given in Appendix 15A and Table 15.5-14; the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable. Doses are based on the dose models in Appendix 15A .

(5) Primary and Secondary side source terms are based on ANSI/ANS–18.1–1984.

Assumptions used for the conservative analysis are the same as the realistic assumptions except the Secondary side source terms at the Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent are assumed.

The steam releases to the atmosphere for the loss of AC power are in Table 15.5-1.

The gamma, beta, and thyroid doses for the loss of AC power to the plant auxiliaries at the exclusion area boundary and low population zone are in Table 15.5-2 for the realistic and conservative analyses. These doses are calculated by the FENCDOSE computer code[16]. The doses for this accident are less than 25 rem whole body, 300 rem beta and 300 rem thyroid. This is well within the limits as defined in 10 CFR 100.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-2. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.2 Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture

Two analyses of the postulated waste gas decay tank rupture are performed:

(1) a realistic analysis, and (2) an analysis based on Regulatory Guide 1.24 (Reference 2). The parameters used for each of these analyses are listed in Table 15.5-3.

The assumptions for the Regulatory Guide analysis are:

(1) The reactor has been operating at full power with 1% defective fuel for the RG 1.24 analysis.

(2) The maximum content of the decay tank assumed to fail is used for the purpose of computing the noble gas inventory in the tank. Radiological decay is taken into account in the computation only for the minimum time period required to transfer the gases from the reactor coolant system to the decay tank. For the Regulatory Guide 1.24 analysis, noble gas and iodine inventories of the tank are given in Table 15.5-4. For the realistic analysis, source terms are based on ANSI/ANS-18.1-1984 methodology[14].

(3) The tank rupture is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank through the Auxiliary Building vent to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that all gas decay tanks will be isolated from each other whenever they are in use.

(4) The short-term (i.e., 0-2 hour) dilution factor at the exclusion area boundary given in Appendix 15A is used to evaluate the doses from the released activity. Doses are based on the dose models presented in Appendix 15A. The gamma, beta, and thyroid doses for the gas decay tank rupture at the exclusion area boundary and low population zone are given in Table 15.5-5 for both the realistic and Regulatory Guide 1.24 analyses.

(5) The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-5. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.3 Environmental Consequences of a Postulated Loss of Coolant Accident

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident do not result in doses which exceed the reference values specified in a 10 CFR 100.

The analysis is based on Regulatory Guide 1.4[3]. The parameters used for this analysis are listed in Table 15.5-6. In addition, an evaluation of the dose to control room operators and an evaluation of the offsite doses resulting from recirculation loop leakage are presented.

Fission Product Release to the Containment

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the emergency core cooling system keeps cladding temperatures well below melting, and limits zirconium-water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the primary containment.

In this analysis, based on Regulatory Guide 1.4[3], a total of 100% of the noble gas core inventory and 25% of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, it is further assumed that 91% is in elemental form, 4% in methyl form, and 5% in particulate form. The core inventory of iodines and noble gases is listed in Table 15.1-5.

Primary Containment Model

The quantity of activity released from the containment was calculated with a single volume model of the containment.

If it is assumed that there are no sources of activity following the initial instantaneous release of fission products to the containment, the equation which describes the time dependent activity or quantity of material in a component is:

$$\frac{dA_{ij}(t)}{dt} = -\Lambda_{ij}A_{ij}(t) + P_{ij}(t) \qquad (1)$$

where A_{ij} is the activity or quantity of material i in component j. P_{ij} is the rate at which activity or material i is added to component j, and Λ_{ij} is the rate at which activity or material i is removed or lost from component j. If both Λ and P are independent of time, then for one material and one component one obtains the solution:

$$A = A_0 e^{-\Lambda t} + \frac{P}{\Lambda}(1 - e^{-\Lambda t}) \qquad (2)$$

where A_o is the initial activity. However, in general, P is time dependent and in some cases Λ is also time dependent.

The addition of material to the component, $P_{ij}(t)$, may come from two sources: (1) flow from another component in the system may add material to the component, (2) material may be produced within the component by radioactive decay. Thus, the addition rate for material i to component j can be expressed as:

$$P_{ij}(t) = P_{ij}^{(1)}(t) + P_{ij}^{(2)}(t) \qquad (3)$$

where:

$P_{ij}^{(1)}(t) = \sum_{jj \neq j}^{n} c_{ijj-j}(t)A_{ijj}(t); c_{ijj-j}(t)$ is the transfer coefficient

of i from component jj to j, and $P_{i}^{(2)}(t) = \sum_{ii}^{n} \Upsilon_{ii-i}A_{iij}(t); \Upsilon_{ii-i}$ is the rate of production

of i from ii in component j. Note that γ_{ii-i} is not normally a function of time or component.

Similarly, the loss from a component can be due to: (1) loss within the component (such as radioactive decay), (2) flow out of the component to other components, and (3) removal from the system. Thus, the loss rate from component j for material i can be expressed as:

$$\Lambda_{ij}(t) = \lambda_i + \Lambda_{ij}^{(2)}(t) + \Lambda_{ij}^{(3)}(t) \qquad (4)$$

where λ_i is the removal rate inside the component due to radioactive decay (neither time nor component dependent),

$$\Lambda_{ij}^{(2)}(t) = \sum_{jj \neq j}^{n} f_{ij-jj}(t); f_{ij-jj}(t)$$ is the transfer coefficient of material i from component j to jj,

and $\Lambda_{ij}^{(3)}(t)$ is the removal from the system.

A computer program Source Transport Program (STP) has been developed to solve equation (1) for each isotope and for two halogen forms (i.e., elemental and or organic). From this, the isotopic concentration airborne in the containment as a function of time and the integrated isotopic leakage from the containment for a given time period can be obtained. Parameters used in the loss-of-coolant accident analysis are listed in Table 15.5-6.

Modeling of Removal Process

For fission products other than iodine, the only removal processes considered are radioactive decay and leakage.

The fission product iodine is assumed to be present in the containment atmosphere in elemental, organic, and particulate form. It is assumed that 91% of the iodine available for leakage from the containment is in elemental (i.e., I_2 vapor) form, 4% is assumed to be in the form of organic iodine compounds (e.g., methyl iodine), and 5% is assumed to be absorbed on airborne particulate matter. In this analysis it was conservatively assumed that the organic form of iodine is not subject to any removal processes other than radioactive decay and leakage from the containment. The elemental and particulate forms of iodine are assumed to behave identically.

The effectiveness of the ice condenser for elemental iodine removal is described in Section 6.5.4. For the calculation of doses, the ice condenser was treated as a time dependent removal process. The time dependent ice condenser iodine removal efficiencies for the Regulatory Guide 1.4 analysis are given in Table 15.5-7.

Ice Condenser

The ice condenser is designed to limit the leakage of airborne activity from the containment in the event of a loss-of-coolant accident. This is accomplished by the removal of heat released to the containment during the accident to the extent necessary to initially maintain that structure below design pressure and then reduce the pressure to near atmospheric. The addition of an alkaline solution such as sodium tetraborate enhances the iodine removal qualities of the melting ice to a point where credit can be assumed in the radiological analyses.

The operation of the containment deck fans (air return fans) is delayed for approximately 10 minutes following a Phase B isolation signal resulting from the loss-of-coolant accident.

This delay in fan operation yields an initial inlet steam-air mixture into the ice condenser of greater than 90% steam by volume which results in more efficient iodine removal by the ice condenser.

As a result of experimental and analytical efforts, the ice condenser system has been proven to be an effective passive system for removing iodine from the containment atmosphere following a loss-of-coolant accident. (Reference 4)

With respect to iodine removal by the ice condenser, the following assumptions were made:

(1) The ice condenser is only effective in removing airborne elemental and particulate iodine from the containment atmosphere.

(2) The ice condenser is modeled as a time dependent removal process.

(3) The ice condenser is no longer effective in removing iodine after all of the ice has been melted using the most conservative assumptions.

Primary Containment Leak Rate

The primary containment leak rate used in the Regulatory Guide 1.4 analysis for the first 24 hours is the design basis leak rate guaranteed in the technical specifications regarding containment leakage and it is 50% of this value for the remainder of the 30 day period. Thus, for the first 24 hours following the accident, the leak rate was assumed to be 0.25% per day and the leak rate was assumed to be 0.125% per day for the remainder of the 30 day period.

The leakage from the primary containment can be grouped into two categories: (1) leakage into the annulus volume and (2) through line leakage to rooms in the Auxiliary Building (see Figure 15.5-1). The environmental effects of the core release source events have been analyzed on the basis that 25% of the total primary containment leakage goes to the Auxiliary Building.

The leakage paths to the Auxiliary Building are tested as part of the normal Appendix J testing of all containment penetrations. An upper bound to leakage to the Auxiliary Building was estimated to be 25% of the total containment leakage. Selecting an upper bound is conservative because an increasing leakage fraction to the Auxiliary Building results in an increasing calculated offsite dose. This upper bound was also selected on the basis that it is large enough to be verified by testing. The periodic Appendix J testing will assure that leakage to the Auxiliary Building remains below 25%. The remaining 75% of the leakage goes to the annulus.

Bypass Leakage Paths

There are no bypass paths for primary containment leakage to go directly to the atmosphere without being filtered. For further details see the discussion on Type E leakage paths in Section 6.2.4.3.1.

Auxiliary Building Release Path

The Auxiliary Building allows holdup and is normally ventilated by the auxiliary building ventilation system. However, upon an ABI signal following a loss-of-coolant accident, the normal ventilation systems to all areas of the Auxiliary Building are shutdown and isolated. Upon Auxiliary Building isolation, the Auxiliary Building gas treatment system (ABGTS) is activated to provide ventilation of the area and filtration of the exhaust to the atmosphere. This system is described in Section 6.2.3.2.3.

Fission products which leak from the primary containment to areas of the Auxiliary Building are diluted in the room atmosphere and travel via ducts and other rooms to the fuel handling area or the waste packaging area where the suctions for the Auxiliary Building gas treatment system are located. The mean holdup time for airborne activity in the Auxiliary Building areas other than the fuel handling area is greater than one hour with the Auxiliary Building isolated and both trains of the ABGTS operating. It has been conservatively assumed in the estimation of activity release that activity leaking to the Auxiliary Building is directly released to the environment for the first four minutes and then through the ABGTS filter system, with a conservatively assumed mean hold-up time of 0.3 hours in the Auxiliary Building before being exhausted. In the Regulatory Guide 1.4 analysis the ABGTS filter system is assumed to have a removal efficiency of 99% for elemental, organic, and particulate iodines. Minor leakage into the ABGTS and EGTS ductwork allows some unfiltered Auxiliary Building air to be released to the environment. This leakage, quantified by testing, is modeled in the LOCA analysis as indicated in Table 15.5-6 and does not significantly impact doses.

The Auxiliary Building internal pressure is maintained at less than atmospheric during normal operation (see Section 9.4.2 and 9.4.3), thereby preventing release to the environment without filtration following a LOCA. The annulus pressure is maintained more negative than the Auxiliary Building internal pressure during normal operation and after a DBA. Therefore, any leakage between the two volumes following a LOCA is into the annulus.

Shield Building Releases

The presence of the annulus between the primary containment and the Shield Building reduces the probability of direct leakage from the vessel to the atmosphere and allows holdup, dilution, sizing, and plate-out of fission products in the Shield Building. The major factor in the effectiveness of the secondary containment is its inherent capability to collect the containment leakage for filtration of the radioactive iodine prior to release to the environment. This effect is greatly enhanced by the recirculation feature of the air handling systems, which forces repeated filtration passes for the major fraction of the primary containment leakage before release to the environment. Seventy-five percent of the primary containment leakage is assumed to go to the annulus volume.

The initial pressure in the annulus is less than atmospheric. However, the dose analysis conservatively assumes the Annulus is at atmospheric pressure at event initiation. After blowdown, the annulus pressure will increase rapidly due to expansion of the containment vessel as a result of primary containment atmosphere temperature and pressure increases. The annulus pressure will continue to rise due to heating of the annulus atmosphere by conduction through the containment vessel. After a delay, the EGTS operates to maintain the annulus pressure below atmospheric pressure.

The EGTS is essentially an annulus recirculation system with pressure activated valves which allow part of the system flow to be exhausted to atmosphere to maintain a "negative" annulus pressure. The system includes absolute and impregnated charcoal filters for removal of halogens. The EGTS combined with ABGTS ensures that all primary containment leakage is filtered before release to the atmosphere.

The EGTS suction in the annulus is located at the top of the containment dome, while nearly all penetrations are located near the bottom of the containment (see Section 6.2), thereby minimizing the probability of leakage directly from the primary containment into the EGTS.

Transfer of activity from the annulus volume to the EGTS suction is assumed to be a statistical process similar mathematically to the decay process, (i.e., the rate of removal from the annulus is proportional to the activity in the annulus). This corresponds an assumption that the activity is homogeneously distributed throughout the mixing volume. Because of the low EGTS flow rate (compared to the annulus volume), the thermal convection due to heating of the containment vessel, and the relative locations of the EGTS suctions (at the top of the dome) and the EGTS recirculation exhausts (at the base of the annulus), a high degree of mixing can be expected. It is conservatively assumed that only 50% of the annulus free volume is available for mixing of activity in the Regulatory Guide 1.4 analysis.

Tables 15.5-8 and 15.5-8A list the EGTS and recirculation flow rates as a function of time after the LOCA, which were used for calculation of activity releases for the Regulatory Guide 1.4 analysis. Table 15.5-8 flow rates are as a result of a postulated single failure loss of one train of EGTS concurrent with LOCA. Table 15.5-8A flow rates are as a result of an alternate single failure scenario resulting in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. Both EGTS fans are in service until operator action is taken to place one fan in standby between one and two hours post accident. The flow path of fission products which are drawn into the air handling systems is shown schematically in Figure 15.5-1 where:

L_0 Represents the flow of activity from primary containment to the annulus

L_1 Represents the flow of activity from primary containment to the Auxiliary Building

L Represents the flow of activity from the annulus into the EGTS

K Represents the ratio of EGTS recirculation flow to total EGTS flow rate

n_f Represents the appropriate filter efficiency

Effectiveness of Double Containment Design

The analysis has demonstrated clearly the benefits of the double containment concept. As would be expected for a double barrier arrangement, the second barrier acts as an effective holdup tank, resulting in substantial reduction in the two-hour inhalation and whole body immersion doses. The expected offsite doses for the 30-day period at the low population zone are also substantially reduced, since the holdup process is effective for the duration of the accident.

The EGTS exhaust flow rate is dependent on the rate of air inleakage to the annulus. In fact, after about 30 minutes following blowdown of the reactor vessel the EGTS exhaust flow is approximately equal to the air inleakage rate. Studies[5] made of leak rates from typical concrete buildings of this type have resulted in leak rates from 4% to 8% per day at a pressure differential of 14 inches of water. Although the pressure differential in this case will be much lower than this value, it has been assumed that a shield building inleakage flow of 250 cfm exists throughout the 30-day period for the single failure scenario which results in loss of one EGTS train concurrent with a LOCA. The inleakage flow for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack (while the other train remains functional) was conservatively assumed to be greater since the resulting annulus pressure is more negative than the original single failure scenario loss of one EGTS train. The long term inleakage flow rates of 832 cfm (until operator action to place one fan in standby) and 604 cfm thereafter are used in the dose analysis. This inleakage flow includes leakage past ventilation system primary containment isolation valves assuming that a single isolation valve fails in the open position.

In order to evaluate the effectiveness of the Shield Building, the following case was analyzed:

50% Mixing Case

At the beginning of the accident, the EGTS starts exhausting filtered fission products to the environs (see Tables 15.5-8 and 15.5-8A). At approximately 114 seconds for the loss of one EGTS train the Annulus pressure becomes less than -0.25 inches w.g. and the effluents are filtered for the duration of the accident. At approximately 60 seconds (for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functonal) the annulus pressure becomes less than minus 0.25 inches w.g., and the effluents are filtered for the duration of the accident. All of the primary containment leakage going to the shield building is assumed to be uniformly mixed in 50% of the annulus free volume.

Emergency Gas Treatment System Filter Efficiencies

The EGTS takes suction from the annulus, and the exhaust gases are drawn through two banks of impregnated charcoal filters in series. Sufficient filter capacity is provided to contain all iodines, inorganic, organic, and particulate available for leakage. Since the air in the annulus is dry, filter efficiencies of greater than 99% are attainable as reported in ORNL-NSIC-4[6]. Heaters and demisters have been incorporated upstream of the filters resulting in a relative humidity of less than 70% in the air entering the filters which further ensures high filter efficiency.

In the Regulatory Guide 1.4 analysis however, an overall removal efficiency of 99% for elemental, organic, and particulate iodine is assumed for the two filter banks in series.

Discussion of Results

The gamma, beta, and thyroid doses for the LOCA at the exclusion area boundary and the low population zone are given in Table 15.5-9. These doses are calculated by the FENCDOSE computer code[16]. The doses are based on the atmospheric dilution factors and dose models given in Appendix 15A. The doses for this accident are less than 25 rem whole body, 300 rem beta, and 300 rem thyroid. The doses are well within the 10 CFR 100 guidelines and reflect the worst case values in consideration of both single failure scenarios.

Loss of Coolant Accident - Environmental Consequences of Recirculation Loop Leakage

Component leakage in the portion of the emergency core cooling system outside containment during the recirculation phase following a loss of coolant accident could result in offsite exposure. The maximum potential leakage for this equipment is specified is Table 6.3-6. This leakage refers to specified design limits for components and normal leakage is expected to be well below those upper limits. Recirculation is assumed in the analysis to start at 10 minutes after the loss of coolant accident. At this time the sump temperature is approximately 160°F (Figure 6.2.1-3). The enthalpy of the sump is approximately 130 BTU/lb. The enthalpy of saturated liquid at 1.0 atmosphere pressure and 212°F is greater than 130 BTU/lb. Therefore, there will be no flashing of the leakage from recirculation loop components, and an iodine partition factor of 0.1 is assumed for the total leakage.

The analysis of the environmental consequences is performed as follows:

Core iodine inventory given in Table 15.1-5 is used. The water volume is comprised of water volumes from the reactor coolant system, accumulators, refueling water storage tank, and ice melt. All the noble gases are assumed to escape to the primary containment. Ninety-seven percent of trituim was assumed to remain liquid and accumulate in the sump, while 3% was assumed to go airborne to the containment. An alternate analysis was also performed assuming 100% of the tritium goes airborne into the containment. Radioactive decay was taken into account in the dose calculation. The major assumptions used in the analysis are listed in Table 15.5-12. The offsite doses at the exclusion area boundary and low population zone for the analysis are given in Table 15.5-13 and reflect the worst case values in consideration of 3% airborne tritium or 100% airborne tritium. The atmospheric dilution factors and dose models discussed in Appendix 15A are used in the dose analysis. The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-13. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

Loss of Coolant Accident - Control Room Operator Doses

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit the whole body gamma dose during an accident period to 5 rem, the thyroid dose to 30 rem and the beta skin dose to 30 rem.

The doses to personnel during a post-accident period originate from several different sources. Exposure within the control room may result from airborne radioactive nuclides entering the control room via the ventilation system. In addition, personnel are exposed to direct gamma radiation penetrating the control room walls, floor, and roof from:

(1) Radioactivity within the primary containment atmosphere

(2) Radioactivity released from containment which may have entered adjacent structures

(3) Radioactivity released from containment which passes above the control room roof

Further exposure of control room personnel to radiation may occur during ingress to the control room from the exclusion area boundary and during egress from the control room to the exclusion area boundary.

In the event of a radioactive release incident, the control room is isolated automatically by a safety injection system signal and/or by radiation signal from beta detectors located in the air intake stream common to the air intake ports at either end of the Control Building. These redundant signals are routed to redundant controls which actuate air-operated isolation dampers downstream of the beta detectors. Operation of the emergency pressurizing fans with inline HEPA filters and charcoal adsorbers is also initiated by these signals. Simultaneously, recirculation air is rerouted automatically through the HEPA filters and charcoal adsorbers. Approximately 711 cfm of outside air, the emergency pressurization air, flows through a duct routed to the emergency recirculation system upstream of the HEPA filters and charcoal adsorbers. This flow of outside air provides the control room with a slight positive pressure relative to the atmosphere outside and to surrounding structures. In addition, the equivalent of 51 cfm of unfiltered outside air enters through the main control room doors and other sources. Isolation dampers located in each intake line may be selectively closed by control room personnel. The selection between the two would be based on the objective of admitting a minimum of airborne activity to the control room via the makeup airflow.

The control room ventilation flow system is shown in Figure 9.4-1.

To evaluate the ability of the control room to meet the requirements of General Design Criterion 19, a time-dependent model of the control room was developed. In this model, the outside air concentration enters the control room via the isolation damper bypass line and the HEPA filters and charcoal absorbers. The concentration in the room is reduced by decay, leakage out, and by recirculation through the HEPA filters and charcoal absorbers. Credit for filtration is taken during two passes through the charcoal absorbers. Using these assumptions, the following equations for the rate of change of the control room concentrations are obtained:

$$\frac{dM}{dt} = C_o(1-K_1)L/V - (L/V)M - \frac{R_c}{V}M - \lambda M \qquad (1)$$

$$\frac{dN}{dt} = \frac{R_c}{V}(1-K_2)\ M - (L/V)N - \lambda N \qquad (2)$$

$$C(t) = M(t) + N(t) \qquad (3)$$

Where:

- $M(t)$ = Once-filtered time-dependent concentration
- $N(t)$ = Twice-filtered (or more) time-dependent concentration
- $C(t)$ = Total time-dependent concentration in control room
- C_o = Concentration of isotope entering air intake
- K_1 = Filter efficiency for a particular isotope during first pass
- K_2 = Filter efficiency for a particular isotope during second pass
- L = Flow rate of outside air into control room and leakage out of control room
- R_c = Recirculated air flow rate through filters
- λ = Decay constant
- V = Control room free volume

These equations are readily solvable if C_o is constant or a simple function of time during a time interval. Since C_o consists of a number of terms involving exponentials, it was assumed to be constant during particular time intervals corresponding to the average concentration during each interval as described below. Solving equations (1), (2), and (3) yields:

$$C(t) = \left[\frac{1-K_1 1-K_2C_o}{W_mV}\right] \times \left[\frac{L}{(1-K_2)}(1-e^{-Wmt}) + \frac{R_cL}{W_nV}(1-e^{-W_nt}) - L(e^{-W_nt} - e^{-W_mt})\right] \qquad (4)$$

Where:

$$W_m = \frac{(L + R_c + \lambda V)}{V}$$

$$W_n = \frac{(L + \lambda V)}{V}$$

The value of C_o used in equation (4) is determined as follows:

$$C_{oi} = (X/Q)_i \frac{\int_{t_i}^{t_{i+1}} R\,dt}{t_{i+1} - t_i} \qquad (5)$$

C_{oi}= Average concentration of activity outside control room during ith time period (Ci/m^3).

$(x/Q)_i$= Atmospheric dilution factor (sec/m^3) during the ith time period.

R= Time dependent release rate of activity from containment (Ci/sec).

The atmospheric dilution factors were determined using the accumulated meteorological data on wind speed, direction, and duration of occurrence obtained from the Watts Bar plant site applied to a building wake dilution model. The dilution factors are calculated by the ARCON96 methodology[8] and are the maximum values for each time period. The worst case is Unit 1 exhaust to intake 2. These factors are applied for the first 8 hours, at which time it is assumed that the operator selects intake 1 which has more favorable dilution factors. The values used in the analysis are given in Table 15.5-14.

Equation (4) is used to determine the concentration at any time within a time period and upon integrating and dividing by the time interval gives the average concentration during the time interval due to inflow of radioactivity with outside air as shown:

$$\overline{C_i} = \int_0^T \frac{C_i(t)dt}{T - 0} \qquad (6)$$

Where:

$T = t - t_{i-1}$

t = Time after accident

t_{i-1} = Time at end of previous time period

Further contributions to the concentration during the time period are due to the concentrations remaining from prior time periods. These contributions are obtained from the following equations:

$$C_{R(i+j)} = M_{R(i+j)} + N_{R(i+j)} \qquad (7)$$

$$\frac{dM_{R(i+j)}}{dt} = -(L/V + R_c/V + \lambda)M_R(i+j) \qquad (8)$$

$$\frac{dN_{R(i+j)}}{dt} = (R_c/V)(1-K_2)M_{R(i+j)} - (L/V + \lambda)N_{R(i+j)} \qquad (9)$$

With initial conditions:

$M_{R(i+j)}(0) = M_{R\,0(i)} =$ (Once-filtered concentration at end of the ith time period.)

$N_{R(i+j)}(0) = N_{R\,0(i)} =$ (Twice-filtered, or more, concentration at end of the ith time period.)

Solving equations (8) and (9) and substituting certain initial condition relations, equation (7) becomes:

$$C_{R(i+j)} = C_{R0(i)}e^{-W_{N}(t-ti)} - M_{R0(i)}K_2(e^{-W_N(t-ti)} - e^{-W_M(t-ti)}) \qquad (10)$$

Integrating equation (10) for each of the prior time periods gives the contribution from these time periods to the present time period. The average concentration is determined for these contributions using the method of equation (6).

Filter efficiencies of 95% for elemental and particulate iodine and 95% for organic iodine were deemed appropriate for the first filter pass. Since the concentrations of iodine in the main control room are such reduced as a result of this filtration, the efficiencies were reduced for the second pass to 70% for elemental and particulate iodine, and 70% for organic iodine.

To account for the unfiltered inleakage, a bypass leak rate (BPR) of 51 cfm was added to the makeup flow (L in equation (1)) of 711 cfm, and the filter factor for the first pass was decreased by the ratio L/(L+BPR). The filter efficiencies for the second pass are not affected by the unfiltered inleakage.

The filter efficiency for noble gases was taken as zero for all cases.

The above equations were incorporated into computer program COROD[17] together with appropriate equations for computing gamma dose, beta dose, and inhalation dose using these average nuclide concentrations and time periods. The whole body gamma dose calculation consists of an incremental volume summation of a point kernel over the control room volume. The principal gammas of each isotope are used to compute the dose from each isotope. The dose computations for beta activity were based on a semi infinite cloud model. Doses to thyroid were based on activity to dose conversion factors. (The equations and various data are given below.) The doses from these calculations are presented in Table 15.5-9. Gamma dose contributions from shine through the control room roof due to the external cloud and from shine through the control room walls from adjacent structures and from containment are computed using an incremental volume summation of a point kernel which includes buildup factors for the concrete shielding. For the calculation of shine through the control room roof, an atmospheric, rectangular volume several thousand feet in height and several control room widths was used. The control room roof is a 2 foot 3-inch-thick concrete slab and is the only shielding considered in this calculation. The average isotope concentrations at the control bay for each time period were used as the source concentrations. For the shine from adjacent structures, the shielding consists of the 3-foot-thick (5 feet in certain areas) control room walls. The doses are calculated similarly to the shine dose through the roof. The average isotope concentrations at the control bay intake for each time period are also used for these calculations.

The shine from the spreading room below the control room is also computed in the same manner as adjacent structures.

Shielding for this computation consists of the 8-inch-thick concrete floor. The summation of the incremental elements is performed over the volume of each room or structure of interest.

In addition to the dose due to shine from surrounding structures and from the passing cloud, the shine from the reactor containment building also contributes to the gamma whole body dose to personnel. This contribution is computed in the same manner as the methods used above. Due to the location of the Auxiliary Building between the Reactor Buildings and the control room and the thicker control room auxiliary building wall near the roof, the minimum ray path through concrete from the containment into the control room below 10 feet above the control floor, is 8 feet. All nuclides released to containment are assumed uniformly distributed and their time-dependent concentrations were used to compute the dose. The dose computed from this source is small.

Several doors penetrate the control room walls, and the dose at these areas would be larger than the doses calculated as described above. The potential shine at these doors and at other penetrations has been evaluated. As a result, hollow steel doors filled with no. 12 lead shot have been incorporated into the design of the shield wall between the control room and the Turbine Building. These doors provide shielding comparable to the concrete walls. Shine through other penetrations was found to be negligible.

Another contribution to the total exposure of control room personnel is the exposure incurred during ingress from and egress to the exclusion area boundary. The doses due to ingress and egress were computed based on the following assumptions:

(1) Five minutes are required to leave the control room and arrive at car or vice versa.

(2) The distance traveled on the access road to the site exclusion boundary is estimated to be 1500 meters. The average car speed is assumed to be 25 mph.

(3) One one-way trip first day, one round-trip/day 2nd through 30th days.

The control room occupancy factors used in this calculation were taken from Murphy and Campe[9]. They are:

100%	occupancy 0-24 hours
60%	occupancy 1-4 days
40%	occupancy 4-30 days.

All atmospheric dilution factors were conservatively based on 5th percentile wind velocity averages.

It was also assumed that initially the makeup air intake would be through the vent admitting the highest radioisotope concentration, but that the main control room personnel would switch intake vents 8 hours after the accident in order to admit a lower amount of airborne activity to the MCR via the makeup air flow.

The whole body, beta, and thyroid doses from the radiation sources discussed above are presented in Table 15.5-9. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose Equations, Data, and Assumptions

The dose from gamma radiation originating within the control room is given by:

$$D_{\Upsilon} = 1.69 \times 10^{4} \sum_{i=1}^{\alpha} \left[\sum_{k=1}^{\beta} TCOT_{ik} \left(\sum_{l=1}^{\Upsilon} \left\{ E_{kl} f_{kl} \left(\frac{\mu_e}{\rho} \right)_l \sum_{m=1}^{\varepsilon} \sum_{n=1}^{\omega} \sum_{q=1}^{\sigma} \frac{\exp(-\mu_{al} \sqrt{x_m^2 + y_n^2 + z_q^2})}{(x_m^2 + y_n^2 + z_q^2)} \cdot \Delta x \Delta y \Delta z \right\} \right) \right] \qquad (11)$$

Where:

$D\gamma$ = Absorbed dose in flesh in mrads

$TCOT_{ik}$=Total concentration integrated over time period i of isotope k in curies/m^3

$E_{k\ell}$ = Energy of gamma ℓ from isotope k in MeV

$f_{k\ell}$ = Number of ℓ gammas of isotope k given off per disintegration

$\left(\frac{\mu_e}{\rho}\right)_\ell$ = Mass attenuation coefficient for flesh determined at the energy of gamma ℓ in cm^2/gram

$\mu_{\alpha\tau}$=Linear attenuation coefficient for air determined at the energy of gamma ℓ in inverse meters

x_m, y_n, z_q=Coordinate distances from the dose point to the source volume element (m,n,q) in meters

$\Delta x, \Delta y, \Delta z$=Dimensions of source element (m,n,q)

α=Number of time periods

β=Number of isotopes

γ=Number of gammas from an isotope

ε=Number of intervals in the x direction

ω=Number of intervals in the y direction

σ=Number of intervals in the z direction

The control room radiation dose from gamma radiation originating outside of the control room and penetrating concrete walls is given as:

$$D_\Upsilon = 1.69 \times 10^4 \sum_{i=1}^{\alpha}\left[\sum_{k=1}^{\beta} C_{o_{ik}}\left(\sum_{l=1}^{\Upsilon}\left\{E_{kl}f_{kl}\left(\frac{\mu_e}{\rho}\right)_l \sum_{m=1}^{\varepsilon}\sum_{n=1}^{\omega}\sum_{q=1}^{\sigma}\frac{\exp(-\mu_{al}\sqrt{x_m^2+y_n^2+z_q^2})}{(x_m^2+y_n^2+z_q^2)}\cdot\exp(-\mu_{cl}t_c\sec\theta)\right.\right.\right.$$

$$\left.\left.\left.\cdot B_c(\mu_{cl}t_c\sec\theta)\cdot\Delta x\Delta y\Delta z\right\}\right)\right](t_i - t_{i-1})$$

Where:

μ_{cl} = Linear attenuation coefficient of concrete determined at the energy of gamma ζ in inverse meters

t_c= Concrete shield thickness in meters

θ = Angle between a vector normal to the shield and a vector from the dose point to the source point

$B_c(\mu_{cl}t_c\sec\theta)$ = Buildup factor for concrete

C_{oik} = Average concentration of isotope k outside the control room during time period i in curies/m^3

t_{i-1}, t_i = Times at the beginning and end of time period i in hours

Other parameters are defined as previously noted.

The dose from beta radiation is given by the semi-infinite cloud immersion dose:

$$D_B = (0.230)\,(X/Q)\left[\sum_{i=1} Q \sum_{k=1} E_{ik}f_{ik}\right] \quad (12)$$

Where:

D_B=Dose due to beta in rem

X/Q=Atmospheric dispersion factor during time period in sec/m^3

Q_i=Accumulated activity release of isotope i during time period

E_{ik}=Average energy of beta k of isotope i

f_{ik}=Number of k betas of isotope i per disintegration

For beta dose in the control room, equation (12) becomes:

$$D_B = (0.230)\sum_{i=1}^{\delta}\sum_{i=1}^{\alpha}\bar{C}_{ij}\sum_{1}^{\beta}E_{ik}f_{ik}(tj - t_{j-1})$$

Where:

$\overline{C}_{ij}$ =Average concentration of isotope i during time period j

Inhalation Dose (Thyroid)

The inhalation dose for a given period of time has the general form:

$$D_I = (X/Q)(B)\left[\sum_{i=1}^{n}(Q_{ij})(DCF_i)\right](t_J - t_{j-1}) \qquad (13)$$

Where:

D_I=Thyroid inhalation dose, rem

X/Q=Site dispersion factor during time period, sec/m^3

B=Breathing rate during time period, m^3/hr

Q_{ij}=Average activity release rate during time period j of iodine isotope i

DCF_i=ICRP-30 Dose conversion factor for iodine isotope i, rem/microcurie inhaled

t_j=Total time at end of period j, hours

For inhalation dose within the control room, equation (13) becomes:

$$D_I = (B)\left[\sum_{i=1}^{n} C_{ij}(DCF_i)\right](t_j - t_{j-1})$$

In this expression C_{ij}, the average concentration of isotope i during time period j, has replaced the following factor:

(X/Q) Q_{ij}

The C_{ij}'s are those determined by equations (4) and (6). The breathing rate factor B, was taken to be 3.47 x 10^{-4} m^3/sec, 1.75 x 10^{-4} m^3/sec, and 2.32 x 10^{-4} m^3/sec for the time intervals of 0-8 hours, 8-24 hours, and 24 hours - 30 days, respectively.

15.5.4 Environmental Consequences of a Postulated Main Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is a leakage from the reactor coolant system to the secondary system in the steam generator. An acceptable primary-to-secondary leakage rate for the main steam line break (MSLB) accident is 1 gallon per minute (gpm) for the faulted steam generator loop and 150 gallons per day (gpd) for each unfaulted steam generator.

A calculation determines the offsite and main control room doses resulting from a MSLB incorporating the above primary-to-secondary criteria. The calculation determined that 1 gpm (at standard temperature and pressure) primary-to-secondary leakage in the faulted steam generator results in site boundary doses within 10CFR100 guidelines and control room doses within the 10CFR50, Appendix A, General Design Criteria (GDC)-19 limit. The calculation uses TVA computer codes STP, FENCDOSE and COROD. The STP output is input to COROD, which determines control room operator dose and FENCDOSE, which determines the 30-day low population zone (LPZ) and the 2-hour exclusion area boundary (EAB) dose.

Two methods for determining the resultant dose for a MSLB in accordance with the Standard Review Plan 15.1.5, Appendix A methodology are:

1. A pre-accident iodine spike where the iodine level in the reactor coolant spiked upward to the maximum allowable limit of 14 μCi/gm I-131 dose equivalent just prior to the initiation of the accident.

2. The reactor coolant at the maximum steady state dose of equivalent I-131 of 0.265 μCi/gm with an accident initiated iodine spike consisting of a 500 times increase on the rate of iodine release from the fuel.

In both cases, the primary-to-secondary side leak is assumed to be 1 gpm in the faulted steam generator loop and 150 gpd in each faulted loop. The primary side activity release uses the Technical Specification (TS) limit design reactor coolant activities, and the secondary side activity uses the Technical Specification limit of $\leq$0.1μCi/gm I-131 dose equivalent.

The steam releases to the atmosphere for the MSLB are in Table 15.5-16.

The gamma, beta and thyroid doses for the MSLB accident at the EAB and LPZ are in Table 15.5-17. The doses from this accident are less than the reference values as listed in 10CFR100 (25 rem whole body and 300 rem thyroid).

The whole body, beta and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-17. The doses are calculated by the COROD computer code. [17] Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] determine the dose. Dose conversion factor in ICRP-30 [25] determine thyroid doses in place of those found in TID-14844.

Assumptions for the MSLB accident:

1. RCS letdown flow of 124.39 gpm.
2. RCS letdown demineralizer efficiency is 1.0 for iodines.
3. ANSI/ASN-18.1-1984 spectrum scaled up to 0.265 or 14 μCi/gm equivalent iodine.
4. Two cases were used. In the first, pre-accident iodine spike of 14 μCi/gm I-131 dose equivalent in the RCS. In the second case, an accident initiated spike which increases the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.
5. Primary side to secondary side leakage of 150 gpd (standard temperature and pressure) per steam generator in the intact loops.
6. The primary-to-secondary leakage mass release to the Environment is 1 gpm (standard temperature and pressure) from the faulted loops.
7. Steam generator secondary inventory released as steam to the atmosphere:

   a) total from the non-defective steam generators (0-2 hrs), 433,079 lbm

   b) total from the non-defective steam generators (2-8 hrs), 870,754 lbm

   c) total from the faulted steam generator (0-30 mins), 96,100 lbm
8. Iodine partition coefficients from steaming of steam generator water:

   i. non-defective steam generators initial inventory and primary-to-secondary leakage, 100.

   ii. faulted steam generator initial inventory and primary-to-secondary leakage, 1.0
9. Atmospheric dilution factors, x/Q, are in Table 15A-2 for offsite and Table 15.5-14 for control room personnel.
10. Main control room related assumptions are in Table 15.5-14.

15.5.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture

Thermal and hydraulic analyses determine the plant response for a design basis steam generator tube rupture (SGTR), and the integrated primary to secondary break flow and mass releases from the ruptured and intact steam generators (SGs) to the condenser and the atmosphere (Section 15.4.3). An analysis of the environmental consequences of the postulated SGTR has also been performed, utilizing the reactor coolant mass and secondary steam mass releases determined in the base thermal and hydraulic analysis (See Reference [38] in Section 15.4). Table 15.5-18 summarizes the parameters used in the SGTR analysis.

The SGTR thermal and hydraulic analysis documents use WBN specific parameters and actual operator performance data, as determined from simulator exercises utilizing the appropriate emergency operating procedures (EOPs). Two cases were analyzed. Case 1: The primary side activity release use the maximum Technical Specification (TS) limit design reactor coolant activities and an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state concentration of 0.265 µCi/gm of I-131 dose equivalent. Case 2: The initial reactor coolant activity is at 14 µCi/gm of I-131 dose equivalent due to a pre-accident iodine spike caused by an RCS transient. For both cases, the secondary side releases uses expected secondary side activities, based on ANSI/ANS-18.1-l984[14] as modified for WBN, and on a 150 gpd/steam generator primary-to-secondary-side leakage. Credit was taken for flashing of the primary coolant (References [34] and [35] of Section 15.4), but "scrubbing" of the iodine in the rising steam bubbles by the water in the steam generator was conservatively neglected. A partition factor of 100 applies to iodine in the remaining unflashed coolant which will boil.

The atmospheric diffusion coefficients (X/Q) for the exclusion area boundary (EAB) and offsite dose determination are the same as those used for the LOCA analysis (Appendix 15A). The X/Q values for the control room operator were determined in the analysis. The LOCA X/Q values are for a release from the shield building vent, whereas the SGTR release is from the top of the main steam valve vault. The methodology for determination of the WBN control room X/Q values is based on computer code ARCON96.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-19. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The gamma, beta, and thyroid dose for the SGTR event are in Table 15.5-19. The doses at the EAB and the low population zone are less than 10% of the 10 CFR 100 limits.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers three cases. The first case is for a Fuel Handling Accident inside containment with the containment closed and the Reactor Building Purge System operating. This analysis is discussed in Section 15.5.6.1 and is based on Regulatory Guide 1.25[11] and NUREG 5009[24]. The second case is for an accident in the spent fuel pool area located in the Auxiliary

Building. This case is discussed in Section 15.5.6.2 and is evaluated using the Alternate Source Term based on Regulatory Guide 1.183[18], "Alternate Source Terms." The third case considered is an open containment case for an accident inside containment where there is open communication between the containment and the Auxiliary Building. This evaluation is discussed in Section 15.5.6.2 and is based on Regulatory Guide 1.183.~~The analysis of a postulated fuel handling accident is based on Regulatory Guide 1.25[11] and NUREG/CR-5009.[24]~~

~~The parameters used for this analysis are listed in Table 15.5-20.~~

~~The bases for the Regulatory Guide 1.25 evaluations are:~~

(1) ~~In the Regulatory Guide 1.25 analysis the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.~~

(2) ~~In the Regulatory Guide 1.25 analysis damage was assumed for all rods in one assembly.~~

(3) ~~The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. In the Regulatory Guide 1.25 analysis, a radial peaking factor of 1.65 is used.~~

(4) ~~For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009[24] as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.~~

(5) ~~Noble gases released to the spent fuel pool are released through the Shield Building vent to the environment.~~

(6) ~~In the Regulatory Guide 1.25 analysis the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).~~

(7) ~~In the Regulatory Guide 1.25 analysis the spent fuel pool decontamination factors for the inorganic and organic iodine are 133 and 1, respectively.~~

(8) ~~All iodine escaping from the pool is exhausted to the environment through charcoal filters.~~

(9) ~~A filter efficiency of 99% is used for elemental and organic iodine for the ABGTS filters and 90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters.~~

(10) ~~No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.~~

(11) ~~The short term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP 30 [25] iodine dose conversion factors.Doses are based on the dose models presented in Appendix 15A.~~

~~The thyroid, gamma, and beta doses for FHAs in the Auxiliary and Reactor Buildings are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10CFR100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated by using Revision 4 of the computer code FENCDOSE[16].~~

~~The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment except that the assigned filter efficiency is 90% for inorganic iodine and 30% for organic iodine since no relative humidity control is provided.~~

~~The radiation dose results of the Regulatory Guide 1.25 fuel handling accident (FHA) is given in Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. For a FHA inside containment when containment penetrations and/or the annulus are open to the Auxiliary Building ABSCE spaces, the containment is isolated by a high radiation signal from monitors in the ventilation system and no credit is assumed for the containment purge filters. The result of a FHA inside the primary containment is well below the limits of 10 CFR 100.~~

~~The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. The gamma and beta doses are based on a one time burn of a TPC fuel element, whereas the thyroid dose is based on a three times burned element. This selection of sources produces higher doses. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.~~

~~Dose equations in TID 14844 [23] were used to determine the dose. Dose conversion factors in ICRP 30 [25] were used to determine thyroid doses in place of those found in TID 14844.~~

15.5.6.1 Fuel Handling Accident Based on Regulatory Guide 1.25

The parameters used for this analysis are listed in Table 15.5-20.

The bases for the Regulatory Guide 1.25 evaluation are:

(1) In the Regulatory Guide 1.25 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.25 analysis damage is assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009 [24] as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.

(5) Noble gases released in the containment are released through the Shield Building vent to the environment.

(6) In the Regulatory Guide 1.25 analysis the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

(7) A filter efficiency of 90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters is used since no relative humidity control is provided.

(8) No credit is taken for natural decay after the activity has been released to the atmosphere.

(9) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.2 Fuel Handling Accident Based on Regulatory Guide 1.183

The analysis of a postulated fuel handling accident in the Auxiliary Building refueling Area is based on Regulatory Guide 1.183. i.e., Alternate Source Terms (AST). The parameters used for this analysis are listed in Table 15.5-20.a.

The bases for evaluation are:

(1) In the Regulatory Guide 1.183 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.183 analysis, damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) The Regulatory Guide 1.183 analysis assumes all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) In the Regulatory Guide 1.183 analysis, the iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) In the Regulatory Guide 1.183 analysis, the overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) In the Regulatory Guide 1.183 analysis, all iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(10) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(11) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.3 Fuel Handling Accident Results

The radiation dose results of the Regulatory Guide 1.25 with the containment closed fuel handling accident (FHA) are given in Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel-handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch-thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch-thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment.

The thyroid, gamma, and beta doses for FHAs for the closed containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10CFR100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated by using Revision 5 of the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel and the thyroid dose is below the limit of 30 rem.

The radiation dose results of the Regulatory Guide 1.183 fuel handling accident (FHA) are given in Table 15.5-23. Alternate source term (AST) described in RG 1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following assumptions are used in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.

- The gap activity is revised to be consistent with that required by RG 1.183.

- The decontamination factors were changed to be consistent with those required by RG. 1.183.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the spent fuel pool to the environment by the ABGTS is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust. In addition, any release from the shield building stack would go through the purge system HEPA and Charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the auxiliary building also isolates and ABGTS is actuated. The release point for ABGTS is the shield building stacks and the releases are filtered through HEPA and Charcoal assemblies. Thus AST analysis for the FHA in the Auxiliary Building that considers no filtration is conservative and an acceptable as the basis for the containment open evaluation.

The thyroid, gamma, and beta doses for FHAs in the Auxiliary and the open containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body and less than the 10 CFR 50.67 limit of 25 rem TEDE. These doses are calculated by using Revision 5 of the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem and the 10 CFR 50.67 limit of 5 rem TEDE.

15.5.7 Environmental Consequences of a Postulated Rod Ejection Accident

This accident is bounded by the loss-of-coolant accident. See Section 15.5.3 for the loss-of-coolant accident.

REFERENCES

(1) Styrikovich, M. A., Martynova, 0. I., Katkovska, K. YA., Dubrovski, I. YA., Smrinova, I. N., "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," translated from Atomnaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

(2) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

(3) Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, June 1974.

(4) D. D. Malinowski, "Iodine Removal in the Ice Condenser System," WCAP-7426, April 1970.

(5) NAA-SR 10100, Conventional Buildings for Reactor Containment.

(6) ORNL-NSIC-4, Behavior of Iodine in Reactor Containment Systems, February 1965.

(7) Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

(8) Ramsdell, J. V. Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes." Prepared by Pacific Northwest laboratory fo the U. S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Revision1, May 1997.

(9) K. G. Murphy and Dr. K. M. Campe "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

(10) Deleted by Amendment 80.

(11) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

(12) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, May 1974.

(13) D. B. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1, December 1971.

(14) ANSI/ANS-18.1-1984, "Radioactive Source Terms for Normal Operations of Light Water Reactors," December 31, 1984.

(15) WCAP-7664, Revision 1, "Radiation Analysis Design Manual-4 Loop Plant," RIMS Number NEB 810126 316, October 1972.

(16) Computer Code FENCDOSE, Code I.D. 262358.

(17) Computer Code COROD, Code I.D. 262347.

(18) Regulatory Guide 1.183 R0, Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, US Nuclear Regulatory Commission, July 2000.~~Not used~~

(19) Not used

(20) NRC Safety Evaluation for Watts Bar Nuclear Plant Unit 1, Amendment 38, for Steam Generator Tubing Voltage Based Alternate Repair Criteria for Outside Diameter Stress Corrosion Cracking (ODSCC) dated February 26, 2002.

(21) NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", dated August 3, 1995.

(22) TVA Letters to NRC "Technical Specification Change No. WBN-TS-99-014 - Steam Generator Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)," dated April 10, 2000, September 18, 2000, August 22, 2001, November 8, 2001 and January 15, 2002.

(23) J.J. Dinunno, et, al "Calculation of Distance Factors for Power and Test Reactor Sites", TIC-14844, March 1962.

(24) NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

(25) International Commission on Radiation Protection (ICRP) Publication 30, Limits for Intakes of Radionuclides by Workers," 1979.

Table 15.5-1 Parameters Used In Loss Of A. C. Power Analyses

	Realistic Analysis	Conservative Analysis
Core thermal power	3565 MWt	3565 MWt
Steam generator tube leak rate prior to and during accident	1 gpm	1.0 gpm
Fuel defects	ANSI/ANS 18.1 - 1984	Technical Specification limit of 0.1 μCi/gm I-131 dose equivalent
Iodine partition factor in steam generator prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	25 gpm	25 gpm
Duration of plant cooldown by secondary system after accident	8 hrs	8 hrs
Steam release from 4 steam generators	444,875 lbm (0-2 hrs) 903,530 lbm (2-8 hrs)	444,875 lbs (0-2 hrs) 903,530 lbs (2-8 hrs)
Meteorology	See Tables 15A-2 & 15.5-14	See Tables 15A-2 & 15.5-14

Table 15.5-2 Doses From Loss Of A/C Power

Conservative Analysis (rem)	**2HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	7.45E-04	4.18E-04	2.10~~2.12~~E-04
Beta	4.48E-04	2.52E-04	2.52~~2.55~~E-03
Thyroid - ICRP-30	4.57E-02	2.57E-02	2.09~~2.11~~E-02

Realistic Analysis (rem)	**2HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	1.80E-08	1.01E-08	5.05~~5.11~~E-09
Beta	1.66E-05	9.29E-06	1.79~~1.81~~E-04
Thyroid - ICRP-30	1.10E-06	6.18E-07	5.03~~5.09~~E-07

Table 15.5-3 Parameters Used In Waste Gas Decay Tank Rupture Analyses

	Realistic Analysis	**Regulatory Guide 1.24 Analysis**
Core thermal power	3565 MWt	3565 MWt
Plant load factor	1.0	1.0
Fuel defects	ANSI/ANS-18.1, 1984	1%
Activity released from GWPS	(1)	See Table 15.5-4
Time of accident	After Tank Fill	At end of equilibrium core cycle
Meteorology	See Table 15.5-14 and Table 15A-2	See Table 15.5-14 and Table 15A-2

(1)Activity based on maximum concentrations of each isotope and actual plant flow rates of the GWPS.

Table 15.5-4 Waste Gas Decay Tank Inventory (One Unit) (Regulatory Guide 1.24 Analysis)

Isotope	Activity (Curies)
Xe-131m	8.9×10^{2}
Xe-133	6.8×10^{4}
Xe-133m	1.0×10^{3}
Xe-135	9.4×10^{2}
Xe-135m	4.8×10^{1}
Xe-137	2.7×10^{-1}
Xe-138	3.2
Kr-83m	1.7×10^{1}
Kr-85	4.2×10^{3}
Kr-85m	1.3×10^{2}
Kr-87	2.9×10^{1}
Kr-88	1.6×10^{2}
Kr-89	1.0×10^{-1}
I-131	4.8×10^{-2}
I-132	---------
I-133	3.3×10^{-2}
I-134	---------
I-135	1.2×10^{-2}

Table 15.5-5 Doses From Gas Decay Tank Rupture

Regulatory Guide 1.24 Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	5.96E-01	1.67E-01	8.43E-01
Beta	1.61E+00	4.51E-01	7.28E+00
Thyroid - ICRP-30	1.29E-02	3.60E-03	6.99E-03

Realistic Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	2.88E-02	8.05E-03	3.81E-02
Beta	1.10E-01	3.08E-02	5.01E-01
Thyroid - ICRP-30	1.21E-02	3.37E-03	6.50E-03

Table 15.5-6 Parameters Used In LOCA Analysis (Page 1 of 2)

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Primary containment free volume	1.27×10^6 ft^3
Annulus free volume	3.75×10^5 ft^3
Primary containment deck (air return) fan flow rate	40,000 cfm
Number of deck (containment air return fans) fans assumed operating	1 of 2
Activity released to primary containment and available for release	
noble gases	100% of core inventory
iodines	25% of core inventory
Form of iodine activity in primary containment available for release	
elemental iodine	91%
methyl iodine	4%
particulate iodine	5%
Ice condenser removal efficiency for elemental and particulate iodine	See Table 15.5-7
Primary containment leak rate (volume percent)	0.25% per day (0-24 hours) 0.125% per day (1-30 days)
Percent of primary containment leakage to auxiliary building	25%
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Delay time of activity in auxiliary building before ABGTS operation	None
Delay time before filtration credit is taken for the ABGTS	4 min
Mean holdup time in auxiliary building after initial 4 minutes	0.3 hours
ABGTS flow rate	9000 cfm

Table 15.5-6 Parameters Used In LOCA Analysis (Page 2 of 2)

Leakage from Auxiliary Building to ABGTS downstream HVAC (bypass of filters)	27.88 cfm
Leakage from ABGTS HVAC into Auxiliary Building	8.87 cfm
Leakage from Auxiliary Building into EGTS downstream HVAC (bypass of filters)	10.7 cfm
Leakage from Auxiliary Building to environment due to single failure of ABGTS (from 30 minutes to 34 minutes post-LOCA)	9900 cfm (for 4 minutes)
Percent of primary containment leakage to annulus	75%
Emergency gas treatment system flow rates	See Table 15.5-8 and Table 15.5-8A
Percent of annulus free volume available for mixing of recirculated activity	50%
Number of emergency gas treatment system air handling units operating	1 of 2
Emergency gas treatment system filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Shield building mixing model (see Section 15.5.3)	50% mixing
Meteorology	See Table 15.5-14 and Table 15A-2.

Table 15.5-7 Ice Condenser Elemental And Particulate Iodine Removal Efficiency[1,2]

Time Interval Post LOCA (Hours)	Iodine Removal Efficiency
0.0 to 0.156	0.96
0.156 to 0.267	0.76
0.267 to 0.323	0.73
0.323 to 0.489	0.71
0.489 to 0.615	0.60
0.615 to 0.768	0.58
0.768 to 0.824	0.40
0.824 to 720	0.0

(1) The ice condenser removal efficiencies given in the above table are used for the Regulatory Guide 1.4 analysis. The inlet steam/air mixture coming into the ice condenser is greater than 90% steam by volume initially due to the delaying of the operation of the containment deck fans. Without the delay of operation of the deck fans, the amount of steam by volume in the inlet mixture initially would be much lower and the ice condenser iodine removal efficiencies would be reduced.

(2) The ice bed iodine removal efficiency, O_I, has been computed on a time dependent basis and is shown in Table 15.5-7. Note that the information presented in Table 15.5-7 has been revised by Westinghouse letter WAT-D-10954. The revised efficiency information is associated with the WCAP-15699, Revision 1 analysis for reduced ice weight. A comparison of the information presented in Table 15.5-7 and the revised information contained in WAT-D-10954 shows that the information in Table 15.5-7 is conservative. Analyses supporting the plant design basis acknowledge the revised efficiency information but shall utilize the information presented in Table.15.5-7.

Table 15.5-8 EMERGENCY GAS TREATMENT SYSTEM FLOW RATES (Sheet 1 of 1)

Time Interval	Time Interval	Recirculation Rate		Exhaust Rate	
(sec)	(hours)	(cfm)	(cfh)	(cfm)	(cfh)
0-30	0-0.0083	0.00	0.00E+00	0.00	0.00E+00
30-39	0.0083-0.0108	3600.00	2.16E+05	0.00	0.00E+00
39-40	0.0108-0.0111	3286.62	1.97E+05	313.38	1.88E+04
40-41	0.0111-0.0114	2352.31	1.41E+05	1247.69	7.49E+04
41-42	0.0114-0.0117	1304.79	7.83E+04	2295.21	1.38E+05
42-43	0.0117-0.0119	362.60	2.18E+04	3237.40	1.94E+05
43-190	0.0119-0.0528	0.00	0.00E+00	3600.00	2.16E+05
190-191	0.0528-0.0531	537.28	3.22E+04	3062.72	1.84E+05
191-192	0.0531-0.0533	733.23	4.40E+04	2866.77	1.72E+05
192-193	0.0533-0.0536	735.14	4.41E+04	2864.86	1.72E+05
193-194	0.0536-0.0539	737.51	4.43E+04	2862.49	1.72E+05
194-199	0.0539-0.0553	745.23	4.47E+04	2854.77	1.71E+05
199-207	0.0553-0.0575	764.12	4.58E+04	2835.89	1.70E+05
207-215	0.0575-0.0597	790.80	4.74E+04	2809.20	1.69E+05
215-225	0.0597-0.0625	825.45	4.95E+04	2774.56	1.66E+05
225-245	0.0625-0.0681	892.72	5.36E+04	2707.29	1.62E+05
245-265	0.0681-0.0736	992.80	5.96E+04	2607.20	1.56E+05
265-285	0.0736-0.0792	1102.40	6.61E+04	2497.61	1.50E+05
285-305	0.0792-0.0847	1217.05	7.30E+04	2382.95	1.43E+05
305-446	0.0847-0.1239	1664.05	9.98E+04	1935.96	1.16E+05
446-601	0.1239-0.1669	2356.72	1.41E+05	1243.29	7.46E+04
601-602	0.1669-0.1672	2661.35	1.60E+05	938.65	5.63E+04
602-1700	0.1672-0.4722	3600.00	2.16E+05	0.00	0.00E+00
1700-1701	0.4722-0.4725	3508.13	2.10E+05	91.87	5.51E+03
1701-1702	0.4725-0.4728	3423.44	2.05E+05	176.56	1.06E+04
1702-1703	0.4728-0.4731	3410.73	2.05E+05	189.27	1.14E+04
1703-1704	0.4731-0.4733	3408.66	2.05E+05	191.34	1.15E+04
1704-1705	0.4733-0.4736	3408.17	2.04E+05	191.83	1.15E+04
1705-1706	0.4736-0.4739	3407.91	2.04E+05	192.09	1.15E+04
1706-1855	0.4739-0.5153	3395.23	2.04E+05	204.77	1.23E+04
1855-2100	0.5153-0.5833	3372.37	2.02E+05	227.64	1.37E+04
2100-30 days*	0.5833-720	3350.00	2.01E+05	250.00	1.50E+04

*Required to maintain annulus pressure when assuming 250 cfm annulus inleakage

Table 15.5-8A Emergency Gas Treatment System Flow Rates (Unit 2)

Time Interval		Time Interval		Recirculation Rate		Exhaust Rate	
(sec)	(sec)	(hours)	(hours)	(cfm)	(cfh)	(cfm)	(cfh)
0	30	0	0.0083	0	0.00E+00	0	0.00E+00
30	39	0.0083	0.0108	7200	4.32E+05	0	0.00E+00
39	40	0.0108	0.0111	6573.24	3.94E+05	626.76	3.76E+04
40	41	0.0111	0.0114	4704.62	2.82E+05	2495.38	1.50E+05
41	42	0.0114	0.0117	2609.58	1.57E+05	4590.42	2.75E+05
42	43	0.0117	0.0119	725.2	4.35E+04	6474.8	3.88E+05
43	71	0.0119	0.0197	0	0.00E+00	7200	4.32E+05
71	78	0.0197	0.0217	0	0.00E+00	7200	4.32E+05
78	79	0.0217	0.0219	1062	6.37E+04	6138	3.68E+05
79	80	0.0219	0.0222	4775	2.87E+05	2425	1.46E+05
80	102	0.0222	0.0283	4337	2.60E+05	2863	1.72E+05
102	132	0.0283	0.0367	4188	2.51E+05	3012	1.81E+05
132	165	0.0367	0.0458	3922	2.35E+05	3278	1.97E+05
165	170	0.0458	0.0472	3762	2.26E+05	3438	2.06E+05
170	210	0.0472	0.0583	3719	2.23E+05	3481	2.09E+05
210	307	0.0583	0.0853	3760	2.26E+05	3440	2.06E+05
307	498	0.0853	0.1383	4050	2.43E+05	3150	1.89E+05
498	602	0.1383	0.1672	4797	2.88E+05	2403	1.44E+05
602	603	0.1672	0.1675	5232	3.14E+05	1968	1.18E+05
603	850	0.1675	0.2361	5137	3.08E+05	1432	8.59E+04
850	1100	0.2361	0.3056	5237	3.14E+05	1332	7.99E+04
1100	1350	0.3056	0.3750	5337	3.20E+05	1232	7.39E+04
1350	1600	0.3750	0.4444	5437	3.26E+05	1132	6.79E+04
1600	1850	0.4444	0.5139	5537	3.32E+05	1032	6.19E+04
1850	2100	0.5139	0.5833	5637	3.38E+05	932	5.59E+04
2100	3600*	0.5833	1.0000	5737	3.44E+05	832	4.99E+04
3600*	30 days	1.0000	30 days	3455	2.07E+05	604	3.62E+04

*Reflects operator action to place one EGTS fan in standby mode at 1 hour.

Table 15.5-9 DOSES FROM LOSS-OF-COOLANT ACCIDENT

(rem)	2Hr EAB	30 Day LPZ	Control Room
Gamma	2.12	2.18	1.05
Beta	1.25	2.61	9.10
Thyroid - ICRP - 30	40.4	14.33	3.75

Breakdown of Control Room Personnel Dose

(rem)	Airborne	Shine	Ingress/Egress	Total
Gamma	1.02	0.005	0.027	1.05
Beta	9.04	0.000	0.060	9.10
Thyroid - ICRP - 30	3.66	0.000	0.090	3.75

Table 15.5-10 Deleted by Amendment 80

Table 15.5-11 Deleted by Amendment 80

Table 15.5-12 PARAMETERS USED IN ANALYSIS OF RECIRCULATION LOOP LEAKAGE FOLLOWING A LOCA

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Recirculation sump water volume	9.63×10^4 ft^3
Activity mixed with recirculation loop water	
Noble gases	0.0
Iodines	50% of core inventory
Tritium	97% to sump (water)
Leakage of ECCS equipment outside containment	See Table 6.3-6
Iodine partition factor for leakage	0.1
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Meteorology	See Table 15.5-14 and Table 15A-2

Table 15.5-13 Doses From Recirculation Loop Leakage Following A LOCA

(rem)	2HR EAB	30 Day LPZ	Control Room
Gamma	4.14E-03	2.28E-02	1.51E-03
Beta	1.36E-03	8.54E-02	1.62E-02
Thyroid - ICRP - 30	1.40E-03	1.53E-01	3.69E-02

Table 15.5-14 Atmospheric Dilution Factors At The Control Building

DILUTION FACTOR (sec/m^3)			
Time Period (hr)	LOCA/FHA	SGTR/MSLB/Loss of AC Power ~~LOSS OF A/C POWER~~	WGDT
0-2	1.09E-03	~~3.85E-03~~ ~~2.61E-03~~2.59E-03	2.56E-03
2-8	9.44E-04	~~3.22E-03~~ ~~2.15E-03~~2.12E-03	~~1.17E-03~~N/A
8-24	1.56E-04*	N/A~~N/A~~	~~7.26E-04~~N/A
24-96	1.16E-04**	N/A~~N/A~~	~~5.21E-04~~N/A
96-720	9.59E-05***	N/A~~N/A~~	~~4.30E-04~~N/A

GENERAL CONTROL ROOM PARAMETERS

Volume	257,198 cu ft
Makeup/pressurization flow	711 cfm
Recirculation flow	2889 cfm
Unfiltered intake	51 cfm
Filter efficiency	95% first pass
	70% second pass
	0% for noble gases, Tritium
Isolation time, T	40 seconds
Occupancy factors:	
0-24 hr	100%
1-4 days	60%
4-30 days	40%

1. All FHA releases are within 2 hours. Thus, only the 0-2 hr X/Q is applicable for the FHA.

* Calculated value for U1 Shield Bldg Vent to East MCR Intake	1.26E-04
** Calculated value for U1 Shield Bldg Vent to East MCR Intake	9.53E-05
*** Calculated value for U1 Shield Bldg Vent to East MCR Intake	8.07E-05

Table 15.5-15 Deleted by Amendment 97

Table 15.5-16 Parameters Used In Steam Line Break Analysis

	Analysis Value
Steam Generator tube leak rate	
Faulted Steam Generator	1.0 gpm
Per Intact Steam Generator	150 gpd
Iodine Partition Factor	
Faulted Steam Generator	1
Intact Steam Generator	100
RCS Letdown flow rate	124.39 gpm
Steam Releases	
Faulted Steam Generator (0-30 minutes)	96,100 lbm
Three Intact Steam Generators (0-2 hrs)	433,079 lbm
Three Intact Steam Generators (2-8 hrs)	870,754 lbm

Table 15.5-17 Doses From Main Steam Line Break

1 gpm Primary-to-Secondary Leakage (ARCON-96 x/Q)	~~Control Room Operator (rem)~~ 2 HR EAB	~~SRP Guidance for GDC 19 Limits (rem)~~30-Day LPZ	~~30 Day LPZ (rem)~~SRP Guidance for 10CFR100 Limits (rem)	Control Room~~2-Hour EAB (Site boundary (rem)~~	SRP Guidance for 10CFR100 Limits (rem)
Pre-Accident Initiated~~Iodine~~ Spike Case (14~~0.265~~ μCi/gm maximum peak~~steady state~~)					
Gamma~~:~~	~~8.10E-03~~2.74E-02	1.11E-02~~5~~	25~~1.25E-01~~	4.32E-03~~1.04E-01~~	~~2.5~~5
Beta~~:~~	~~6.52E-02~~8.80E-03	4.20E-03~~30~~	300~~3.02E-02~~	3.96E-02~~2.54E-02~~	30
Thyroid - ICRP-30~~Inhalation (ICRP-30):~~	~~10.4E+00~~2.41E+00	1.21E+00~~30~~	300~~4.78E+00~~	7.38E+00~~3.09E+00~~	30
~~Pre-~~Accident Initiated~~Iodine~~ Spike Case (0.265~~14~~ μCi/gm steady state~~max peak~~)					
Gamma~~:~~	~~4.36E-03~~1.04E-01	~~5~~1.25E-01	~~1.11E-02~~2.5	~~2.74E-02~~8.00E-03	~~25~~5
Beta~~:~~	~~4.00E-02~~2.54E-02	~~30~~3.02E-02	~~4.20E-03~~30	~~8.80E-03~~6.44E-02	300
Thyroid - ICRP-30~~Inhalation (ICRP-30):~~	~~7.44E+00~~3.09E+00	~~30~~4.78E+00	~~1.21E+00~~30	~~2.41E+00~~1.03E+01	300

Table 15.5-18 Parameters Used In Steam Generator Tube Rupture Analysis

Primary Side Activity	Technical Specification Limit
Secondary Side Activity	ANSI/ANS-18.1-1984 (Expected levels, 150 gpd/SG)
Iodine Spiking Factor	Case 1: Accident initiated spike of 500 times equilibrium iodine concentration Case 2: Pre-accident spike of 14 μCi/gm I-131 dose equivalent
Iodine Partition Factor	100
Secondary Side Mass Release (Ruptured Steam Generator) 0 -2 hours 2 - 8 hours	 103,300 lbm 32,800 lbm
Secondary Side Mass Release (Intact Steam Generator) 0 - 2 hours 2 - 8 hours	 492,100 lbm 900,200 lbm
Primary Coolant Mass Release (Total) 0 - 2 hours	 191,400 lbm
Primary Coolant Mass Release (Flashed) 0 - 2 hours	 10,077.2 lbm
~~Atmospheric diffusion coefficients for control room Operator doses~~Meteorology	~~2.61×10^{-3} (0 - 2 hrs)~~ ~~2.15×10^{-3} (2 - 8 hrs)~~See Table 15A-2 and 15.5-14

Table 15.5-19 Doses From Steam Generator Tube Rupture

Pre-Accident Initiated Spike Case (14 µCi/gm maximum peak)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	3.78E-01	1.11E-01	~~6.28~~6.22E-02
Beta	2.26E-01	6.92E-02	~~7.07~~7.01E-01
Thyroid - ICRP-30	1.39E+01	3.79E+00	~~1.24~~1.23E+01

Accident Initiated Iodine Spike Case (0.265 µCi/gm steady state)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	5.46E-01	1.60E-01	~~5.76~~5.71E-02
Beta	2.51E-01	7.73E-02	~~6.70~~6.64E-01
Thyroid - ICRP-30	7.19E+00	2.12E+00	~~2.03~~2.01E+00

Table 15.5-20 Parameters Used In Fuel Handling Accident Analysis

Regulatory Guide 1.25 Analysis		
Time between plant shutdown and accident	100 hours	
Damage to fuel assembly	All rods ruptured	
Fuel assembly activity	Highest powered fuel assembly in core region discharged	
Activity release to spent fuel pool	Gap activity in ruptured rods(1)	
Radial peaking factor	1.65	
Form of iodine activity released~~ to spent fuel pool~~		
elemental iodine	99.75%	
methyl iodine	0.25%	
~~Decontamination factor in spent fuel pool~~		
~~elemental iodine~~	~~133~~	
~~methyl iodine~~	~~1~~	
~~noble gases~~	~~1~~	
Filter efficiencies in auxiliary building	~~ABGTS(2)~~	RBPVS(~~3~~2)
elemental iodine	~~99%~~	90%
methyl iodine	~~99%~~	30%
Amount of mixing of activity in Auxiliary Building	None	
Meteorology	See Table 15.5-14 and Table 15A-2	

(1) 10% of the total radioactive iodine except for 12% of I-131 and 10% of total noble gases, except for 14% for Kr-85, 5% for Xe-133 and 2% for Xe-135 in the damaged rods at the time of the accident.

~~(2) Auxiliary Building Gas Treatment System~~

(~~3~~2) Reactor Building Purge Ventilation System

Table 15.5-20a Parameters Used In Fuel Handling Accident Analysis

Regulatory Guide 1.183 Analysis	
Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool	
elemental iodine	99.85%(AST)
methyl iodine	0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology	See Table 15.5-14 and Table15A-2

(1) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Table 15.5-21 Nuclear Characteristics Of Highest Rated Discharged Assembly Used In The Analysis

Core thermal power	3565 MWt
Number of assemblies	193
Fuel rods per assembly	264
Core average assembly power	18.47 MWt
Discharged Assembly	
Radial peak to average ratio	1.65

Table 15.5-22 Deleted by Amendment 80

Table 15.5-23
Doses From A Fuel Handling Accident ~~Regulatory Guide 1.25 Analysis Doses From A Fuel Handling Accident~~ (FHA) (rem)

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Auxiliary Building (rem) or In Containment - Containment Open (rem)

~~(rem)~~	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	~~3.994E-01~~4.29E-01	~~9.278E-02~~1.20E-01	~~4.935E-01~~5.86E-01
Beta	~~1.177E+00~~1.19E+00	~~2.734E-01~~3.33E-01	~~4.068E+00~~4.68E+00
Thyroid - ICRP-30	~~1.577E+00~~5.51E+01	~~3.663E-01~~1.54E+01	~~1.540E+00~~1.32E+01
TEDE	2.38E+00	6.66E-01	1.02E-00

Doses from Fuel Handling Accident Regulatory Guide 1.25 Analyses

FHA in Reactor Building, Containment Closed (rem)

~~(rem)~~	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	~~4.102E-01~~4.31E-01	~~9.529E-02~~1.21E-01	~~2.677E-01~~2.72E-01
Beta	~~1.182E+00~~1.24E+00	~~2.746E-01~~3.48E-01	~~2.207E+00~~2.25E+00
Thyroid - ICRP-30	~~39.42E+00~~4.15E+01	~~9.158E+00~~1.16E+01	~~5.209E+00~~6.81E+00

~~**FHA in Reactor Building With Containment Penetrations Open to Auxiliary Building**~~

~~(rem)~~	~~2 HR EAB~~	~~30 DAY LPZ~~	~~CONTROL ROOM~~
~~Gamma~~	~~4.037E-01~~	~~9.378E-02~~	~~5.042E-01~~
~~Beta~~	~~1.189E+00~~	~~2.761E-01~~	~~4.156E+00~~
~~Thyroid - ICRP-30~~	~~3.113E+00~~	~~7.230E-01~~	~~6.436E+00~~

Table 15.5-24 Deleted by Amendment 80

Table 15.5-25 Deleted by Amendment 80

Enclosure 3

WBN Unit 2 – Revised FSAR Section 15.5

15.5 ENVIRONMENTAL CONSEQUENCES OF ACCIDENTS

15.5.1 Environmental Consequences of a Postulated Loss of AC Power to the Plant Auxiliaries

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is leakage from the reactor coolant system (RCS) to the secondary system in the steam generator. A conservative analysis of the potential offsite doses resulting from this accident is presented with steam generator leakage as a parameter. This analysis incorporates assumptions of a Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent, and a realistic source term. Parameters used in both the realistic and conservative analyses are listed in Table 15.5-1.

The realistic assumptions that determine the equilibrium concentrations of isotopes in the secondary system are as follows:

(1) Primary coolant activity is associated with 0.125% defective fuel and is given in Table 11.1-7.

(2) The iodine partition factor in the steam generators is:

$$\frac{\text{amount of iodine/unit mass steam}}{\text{amount of iodine/unit mass liquid}} = 0.01$$

(3) No noble gas is dissolved or contained in the steam generator water, i.e., all noble gas leaked to the secondary system is continuously released with steam from the steam generators through the condenser off gas system.

(4) The 0-2 and 2-8 hour atmospheric dilution factors given in Appendix 15A and Table 15.5-14; the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable. Doses are based on the dose models in Appendix 15A .

(5) Primary and Secondary side source terms are based on ANSI/ANS–18.1–1984.

Assumptions used for the conservative analysis are the same as the realistic assumptions except the Secondary side source terms at the Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent are assumed.

The steam releases to the atmosphere for the loss of AC power are in Table 15.5-1.

The gamma, beta, and thyroid doses for the loss of AC power to the plant auxiliaries at the exclusion area boundary and low population zone are in Table 15.5-2 for the realistic and conservative analyses. These doses are calculated by the FENCDOSE computer code[16]. The doses for this accident are less than 25 rem whole body, 300 rem beta and 300 rem thyroid. This is well within the limits as defined in 10 CFR 100.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-2. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.2 Environmental Consequences of a Postulated Waste Gas Decay Tank Rupture

Two analyses of the postulated waste gas decay tank rupture are performed:

(1) a realistic analysis, and (2) an analysis based on Regulatory Guide 1.24 (Reference 2). The parameters used for each of these analyses are listed in Table 15.5-3.

The assumptions for the Regulatory Guide analysis are:

(1) The reactor has been operating at full power with 1% defective fuel for the RG 1.24 analysis.

(2) The maximum content of the decay tank assumed to fail is used for the purpose of computing the noble gas inventory in the tank. Radiological decay is taken into account in the computation only for the minimum time period required to transfer the gases from the reactor coolant system to the decay tank. For the Regulatory Guide 1.24 analysis, noble gas and iodine inventories of the tank are given in Table 15.5-4. For the realistic analysis, source terms are based on ANSI/ANS-18.1-1984 methodology[14].

(3) The tank rupture is assumed to occur immediately upon completion of the waste gas transfer, releasing the entire contents of the tank through the Auxiliary Building vent to the outside atmosphere. The assumption of the release of the noble gas inventory from only a single tank is based on the fact that all gas decay tanks will be isolated from each other whenever they are in use.

(4) The short-term (i.e., 0-2 hour) dilution factor at the exclusion area boundary given in Appendix 15A is used to evaluate the doses from the released activity. Doses are based on the dose models presented in Appendix 15A. The gamma, beta, and thyroid doses for the gas decay tank rupture at the exclusion area boundary and low population zone are given in Table 15.5-5 for both the realistic and Regulatory Guide 1.24 analyses.

(5) The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-5. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

15.5.3 Environmental Consequences of a Postulated Loss of Coolant Accident

The results of the analysis presented in this section demonstrate that the amounts of radioactivity released to the environment in the event of a loss-of-coolant accident do not result in doses which exceed the reference values specified in a 10 CFR 100.

The analysis is based on Regulatory Guide 1.4[3]. The parameters used for this analysis are listed in Table 15.5-6. In addition, an evaluation of the dose to control room operators and an evaluation of the offsite doses resulting from recirculation loop leakage are presented.

Fission Product Release to the Containment

Following a postulated double-ended rupture of a reactor coolant pipe with subsequent blowdown, the emergency core cooling system keeps cladding temperatures well below melting, and limits zirconium-water reactions to an insignificant level, assuring that the core remains intact and in place. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the primary containment.

In this analysis, based on Regulatory Guide 1.4[3], a total of 100% of the noble gas core inventory and 25% of the core iodine inventory is assumed to be immediately available for leakage from the primary containment. Of the halogen activity available for release, it is further assumed that 91% is in elemental form, 4% in methyl form, and 5% in particulate form. The core inventory of iodines and noble gases is listed in Table 15.1-5.

Primary Containment Model

The quantity of activity released from the containment was calculated with a single volume model of the containment.

If it is assumed that there are no sources of activity following the initial instantaneous release of fission products to the containment, the equation which describes the time dependent activity or quantity of material in a component is:

$$\frac{dA_{ij}(t)}{dt} = -\Lambda_{ij}A_{ij}(t) + P_{ij}(t) \qquad (1)$$

where A_{ij} is the activity or quantity of material i in component j. P_{ij} is the rate at which activity or material i is added to component j, and Λ_{ij} is the rate at which activity or material i is removed or lost from component j. If both Λ and P are independent of time, then for one material and one component one obtains the solution:

$$A = A_0 e^{-\Lambda t} + \frac{P}{\Lambda}(1 - e^{-\Lambda t}) \qquad (2)$$

where A_o is the initial activity. However, in general, P is time dependent and in some cases Λ is also time dependent.

The addition of material to the component, $P_{ij}(t)$, may come from two sources: (1) flow from another component in the system may add material to the component, (2) material may be produced within the component by radioactive decay. Thus, the addition rate for material i to component j can be expressed as:

$$P_{ij}(t) = P_{ij}^{(1)}(t) + P_{ij}^{(2)}(t) \qquad (3)$$

where:

$P_{ij}^{(1)}(t) = \sum_{jj \neq j}^{n} c_{ijj-j}(t)A_{ijj}(t); c_{ijj-j}(t)$ is the transfer coefficient

of i from component jj to j, and $P_{i}^{(2)}(t) = \sum_{ii}^{n} \Upsilon_{ii-i}A_{iij}(t); \Upsilon_{ii-i}$ is the rate of production

of i from ii in component j. Note that γ_{ii-i} is not normally a function of time or component.

Similarly, the loss from a component can be due to: (1) loss within the component (such as radioactive decay), (2) flow out of the component to other components, and (3) removal from the system. Thus, the loss rate from component j for material i can be expressed as:

$$\Lambda_{ij}(t) = \lambda_i + \Lambda_{ij}^{(2)}(t) + \Lambda_{ij}^{(3)}(t) \qquad (4)$$

where λ_i is the removal rate inside the component due to radioactive decay (neither time nor component dependent),

$$\Lambda_{ij}^{(2)}(t) = \sum_{jj \neq j}^{n} f_{ij-jj}(t); f_{ij-jj}(t)$$ is the transfer coefficient of material i from component j to jj,

and $\Lambda_{ij}^{(3)}(t)$ is the removal from the system.

A computer program Source Transport Program (STP) has been developed to solve equation (1) for each isotope and for two halogen forms (i.e., elemental and or organic). From this, the isotopic concentration airborne in the containment as a function of time and the integrated isotopic leakage from the containment for a given time period can be obtained. Parameters used in the loss-of-coolant accident analysis are listed in Table 15.5-6.

Modeling of Removal Process

For fission products other than iodine, the only removal processes considered are radioactive decay and leakage.

The fission product iodine is assumed to be present in the containment atmosphere in elemental, organic, and particulate form. It is assumed that 91% of the iodine available for leakage from the containment is in elemental (i.e., I_2 vapor) form, 4% is assumed to be in the form of organic iodine compounds (e.g., methyl iodine), and 5% is assumed to be absorbed on airborne particulate matter. In this analysis it was conservatively assumed that the organic form of iodine is not subject to any removal processes other than radioactive decay and leakage from the containment. The elemental and particulate forms of iodine are assumed to behave identically.

The effectiveness of the ice condenser for elemental iodine removal is described in Section 6.5.4. For the calculation of doses, the ice condenser was treated as a time dependent removal process. The time dependent ice condenser iodine removal efficiencies for the Regulatory Guide 1.4 analysis are given in Table 15.5-7.

Ice Condenser

The ice condenser is designed to limit the leakage of airborne activity from the containment in the event of a loss-of-coolant accident. This is accomplished by the removal of heat released to the containment during the accident to the extent necessary to initially maintain that structure below design pressure and then reduce the pressure to near atmospheric. The addition of an alkaline solution such as sodium tetraborate enhances the iodine removal qualities of the melting ice to a point where credit can be assumed in the radiological analyses.

The operation of the containment deck fans (air return fans) is delayed for approximately 10 minutes following a Phase B isolation signal resulting from the loss-of-coolant accident.

This delay in fan operation yields an initial inlet steam-air mixture into the ice condenser of greater than 90% steam by volume which results in more efficient iodine removal by the ice condenser.

As a result of experimental and analytical efforts, the ice condenser system has been proven to be an effective passive system for removing iodine from the containment atmosphere following a loss-of-coolant accident. (Reference 4)

With respect to iodine removal by the ice condenser, the following assumptions were made:

(1) The ice condenser is only effective in removing airborne elemental and particulate iodine from the containment atmosphere.

(2) The ice condenser is modeled as a time dependent removal process.

(3) The ice condenser is no longer effective in removing iodine after all of the ice has been melted using the most conservative assumptions.

Primary Containment Leak Rate

The primary containment leak rate used in the Regulatory Guide 1.4 analysis for the first 24 hours is the design basis leak rate guaranteed in the technical specifications regarding containment leakage and it is 50% of this value for the remainder of the 30 day period. Thus, for the first 24 hours following the accident, the leak rate was assumed to be 0.25% per day and the leak rate was assumed to be 0.125% per day for the remainder of the 30 day period.

The leakage from the primary containment can be grouped into two categories: (1) leakage into the annulus volume and (2) through line leakage to rooms in the Auxiliary Building (see Figure 15.5-1). The environmental effects of the core release source events have been analyzed on the basis that 25% of the total primary containment leakage goes to the Auxiliary Building.

The leakage paths to the Auxiliary Building are tested as part of the normal Appendix J testing of all containment penetrations. An upper bound to leakage to the Auxiliary Building was estimated to be 25% of the total containment leakage. Selecting an upper bound is conservative because an increasing leakage fraction to the Auxiliary Building results in an increasing calculated offsite dose. This upper bound was also selected on the basis that it is large enough to be verified by testing. The periodic Appendix J testing will assure that leakage to the Auxiliary Building remains below 25%. The remaining 75% of the leakage goes to the annulus.

Bypass Leakage Paths

There are no bypass paths for primary containment leakage to go directly to the atmosphere without being filtered. For further details see the discussion on Type E leakage paths in Section 6.2.4.3.1.

Auxiliary Building Release Path

The Auxiliary Building allows holdup and is normally ventilated by the auxiliary building ventilation system. However, upon an ABI signal following a loss-of-coolant accident, the normal ventilation systems to all areas of the Auxiliary Building are shutdown and isolated. Upon Auxiliary Building isolation, the Auxiliary Building gas treatment system (ABGTS) is activated to provide ventilation of the area and filtration of the exhaust to the atmosphere. This system is described in Section 6.2.3.2.3.

Fission products which leak from the primary containment to areas of the Auxiliary Building are diluted in the room atmosphere and travel via ducts and other rooms to the fuel handling area or the waste packaging area where the suctions for the Auxiliary Building gas treatment system are located. The mean holdup time for airborne activity in the Auxiliary Building areas other than the fuel handling area is greater than one hour with the Auxiliary Building isolated and both trains of the ABGTS operating. It has been conservatively assumed in the estimation of activity release that activity leaking to the Auxiliary Building is directly released to the environment for the first four minutes and then through the ABGTS filter system, with a conservatively assumed mean hold-up time of 0.3 hours in the Auxiliary Building before being exhausted. In the Regulatory Guide 1.4 analysis the ABGTS filter system is assumed to have a removal efficiency of 99% for elemental, organic, and particulate iodines. Minor leakage into the ABGTS and EGTS ductwork allows some unfiltered Auxiliary Building air to be released to the environment. This leakage, quantified by testing, is modeled in the LOCA analysis as indicated in Table 15.5-6 and does not significantly impact doses.

The Auxiliary Building internal pressure is maintained at less than atmospheric during normal operation (see Section 9.4.2 and 9.4.3), thereby preventing release to the environment without filtration following a LOCA. The annulus pressure is maintained more negative than the Auxiliary Building internal pressure during normal operation and after a DBA. Therefore, any leakage between the two volumes following a LOCA is into the annulus.

Shield Building Releases

The presence of the annulus between the primary containment and the Shield Building reduces the probability of direct leakage from the vessel to the atmosphere and allows holdup, dilution, sizing, and plate-out of fission products in the Shield Building. The major factor in the effectiveness of the secondary containment is its inherent capability to collect the containment leakage for filtration of the radioactive iodine prior to release to the environment. This effect is greatly enhanced by the recirculation feature of the air handling systems, which forces repeated filtration passes for the major fraction of the primary containment leakage before release to the environment. Seventy-five percent of the primary containment leakage is assumed to go to the annulus volume.

The initial pressure in the annulus is less than atmospheric. However, the dose analysis conservatively assumes the Annulus is at atmospheric pressure at event initiation. After blowdown, the annulus pressure will increase rapidly due to expansion of the containment vessel as a result of primary containment atmosphere temperature and pressure increases. The annulus pressure will continue to rise due to heating of the annulus atmosphere by conduction through the containment vessel. After a delay, the EGTS operates to maintain the annulus pressure below atmospheric pressure.

The EGTS is essentially an annulus recirculation system with pressure activated valves which allow part of the system flow to be exhausted to atmosphere to maintain a "negative" annulus pressure. The system includes absolute and impregnated charcoal filters for removal of halogens. The EGTS combined with ABGTS ensures that all primary containment leakage is filtered before release to the atmosphere.

The EGTS suction in the annulus is located at the top of the containment dome, while nearly all penetrations are located near the bottom of the containment (see Section 6.2), thereby minimizing the probability of leakage directly from the primary containment into the EGTS.

Transfer of activity from the annulus volume to the EGTS suction is assumed to be a statistical process similar mathematically to the decay process, (i.e., the rate of removal from the annulus is proportional to the activity in the annulus). This corresponds an assumption that the activity is homogeneously distributed throughout the mixing volume. Because of the low EGTS flow rate (compared to the annulus volume), the thermal convection due to heating of the containment vessel, and the relative locations of the EGTS suctions (at the top of the dome) and the EGTS recirculation exhausts (at the base of the annulus), a high degree of mixing can be expected. It is conservatively assumed that only 50% of the annulus free volume is available for mixing of activity in the Regulatory Guide 1.4 analysis.

Tables 15.5-8 and 15.5-8A list the EGTS and recirculation flow rates as a function of time after the LOCA, which were used for calculation of activity releases for the Regulatory Guide 1.4 analysis. Table 15.5-8 flow rates are as a result of a postulated single failure loss of one train of EGTS concurrent with LOCA. Table 15.5-8A flow rates are as a result of an alternate single failure scenario resulting in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functional. Both EGTS fans are in service until operator action is taken to place one fan in standby between one and two hours post accident. The flow path of fission products which are drawn into the air handling systems is shown schematically in Figure 15.5-1 where:

L_0 Represents the flow of activity from primary containment to the annulus

L_1 Represents the flow of activity from primary containment to the Auxiliary Building

L Represents the flow of activity from the annulus into the EGTS

K Represents the ratio of EGTS recirculation flow to total EGTS flow rate

n_f Represents the appropriate filter efficiency

Effectiveness of Double Containment Design

The analysis has demonstrated clearly the benefits of the double containment concept. As would be expected for a double barrier arrangement, the second barrier acts as an effective holdup tank, resulting in substantial reduction in the two-hour inhalation and whole body immersion doses. The expected offsite doses for the 30-day period at the low population zone are also substantially reduced, since the holdup process is effective for the duration of the accident.

The EGTS exhaust flow rate is dependent on the rate of air inleakage to the annulus. In fact, after about 30 minutes following blowdown of the reactor vessel the EGTS exhaust flow is approximately equal to the air inleakage rate. Studies[5] made of leak rates from typical concrete buildings of this type have resulted in leak rates from 4% to 8% per day at a pressure differential of 14 inches of water. Although the pressure differential in this case will be much lower than this value, it has been assumed that a shield building inleakage flow of 250 cfm exists throughout the 30-day period for the single failure scenario which results in loss of one EGTS train concurrent with a LOCA. The inleakage flow for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack (while the other train remains functional) was conservatively assumed to be greater since the resulting annulus pressure is more negative than the original single failure scenario loss of one EGTS train. The long term inleakage flow rates of 832 cfm (until operator action to place one fan in standby) and 604 cfm thereafter are used in the dose analysis. This inleakage flow includes leakage past ventilation system primary containment isolation valves assuming that a single isolation valve fails in the open position.

In order to evaluate the effectiveness of the Shield Building, the following case was analyzed:

50% Mixing Case

At the beginning of the accident, the EGTS starts exhausting filtered fission products to the environs (see Tables 15.5-8 and 15.5-8A). At approximately 114 seconds for the loss of one EGTS train the Annulus pressure becomes less than -0.25 inches w.g. and the effluents are filtered for the duration of the accident. At approximately 60 seconds (for the single failure scenario which results in one pressure control train in full exhaust to the shield building exhaust stack while the other train remains functonal) the annulus pressure becomes less than minus 0.25 inches w.g., and the effluents are filtered for the duration of the accident. All of the primary containment leakage going to the shield building is assumed to be uniformly mixed in 50% of the annulus free volume.

Emergency Gas Treatment System Filter Efficiencies

The EGTS takes suction from the annulus, and the exhaust gases are drawn through two banks of impregnated charcoal filters in series. Sufficient filter capacity is provided to contain all iodines, inorganic, organic, and particulate available for leakage. Since the air in the annulus is dry, filter efficiencies of greater than 99% are attainable as reported in ORNL-NSIC-4[6]. Heaters and demisters have been incorporated upstream of the filters resulting in a relative humidity of less than 70% in the air entering the filters which further ensures high filter efficiency.

In the Regulatory Guide 1.4 analysis however, an overall removal efficiency of 99% for elemental, organic, and particulate iodine is assumed for the two filter banks in series.

Discussion of Results

The gamma, beta, and thyroid doses for the LOCA at the exclusion area boundary and the low population zone are given in Table 15.5-9. These doses are calculated by the FENCDOSE computer code[16]. The doses are based on the atmospheric dilution factors and dose models given in Appendix 15A. The doses for this accident are less than 25 rem whole body, 300 rem beta, and 300 rem thyroid. The doses are well within the 10 CFR 100 guidelines and reflect the worst case values in consideration of both single failure scenarios.

Loss of Coolant Accident - Environmental Consequences of Recirculation Loop Leakage

Component leakage in the portion of the emergency core cooling system outside containment during the recirculation phase following a loss of coolant accident could result in offsite exposure. The maximum potential leakage for this equipment is specified is Table 6.3-6. This leakage refers to specified design limits for components and normal leakage is expected to be well below those upper limits. Recirculation is assumed in the analysis to start at 10 minutes after the loss of coolant accident. At this time the sump temperature is approximately 160°F (Figure 6.2.1-3). The enthalpy of the sump is approximately 130 BTU/lb. The enthalpy of saturated liquid at 1.0 atmosphere pressure and 212°F is greater than 130 BTU/lb. Therefore, there will be no flashing of the leakage from recirculation loop components, and an iodine partition factor of 0.1 is assumed for the total leakage.

The analysis of the environmental consequences is performed as follows:

Core iodine inventory given in Table 15.1-5 is used. The water volume is comprised of water volumes from the reactor coolant system, accumulators, refueling water storage tank, and ice melt. All the noble gases are assumed to escape to the primary containment. Ninety-seven percent of trituim was assumed to remain liquid and accumulate in the sump, while 3% was assumed to go airborne to the containment. An alternate analysis was also performed assuming 100% of the tritium goes airborne into the containment. Radioactive decay was taken into account in the dose calculation. The major assumptions used in the analysis are listed in Table 15.5-12. The offsite doses at the exclusion area boundary and low population zone for the analysis are given in Table 15.5-13 and reflect the worst case values in consideration of 3% airborne tritium or 100% airborne tritium. The atmospheric dilution factors and dose models discussed in Appendix 15A are used in the dose analysis. The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-13. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

Loss of Coolant Accident - Control Room Operator Doses

In accordance with General Design Criterion 19, the control room ventilation system and shielding have been designed to limit the whole body gamma dose during an accident period to 5 rem, the thyroid dose to 30 rem and the beta skin dose to 30 rem.

The doses to personnel during a post-accident period originate from several different sources. Exposure within the control room may result from airborne radioactive nuclides entering the control room via the ventilation system. In addition, personnel are exposed to direct gamma radiation penetrating the control room walls, floor, and roof from:

(1) Radioactivity within the primary containment atmosphere

(2) Radioactivity released from containment which may have entered adjacent structures

(3) Radioactivity released from containment which passes above the control room roof

Further exposure of control room personnel to radiation may occur during ingress to the control room from the exclusion area boundary and during egress from the control room to the exclusion area boundary.

In the event of a radioactive release incident, the control room is isolated automatically by a safety injection system signal and/or by radiation signal from beta detectors located in the air intake stream common to the air intake ports at either end of the Control Building. These redundant signals are routed to redundant controls which actuate air-operated isolation dampers downstream of the beta detectors. Operation of the emergency pressurizing fans with inline HEPA filters and charcoal adsorbers is also initiated by these signals. Simultaneously, recirculation air is rerouted automatically through the HEPA filters and charcoal adsorbers. Approximately 711 cfm of outside air, the emergency pressurization air, flows through a duct routed to the emergency recirculation system upstream of the HEPA filters and charcoal adsorbers. This flow of outside air provides the control room with a slight positive pressure relative to the atmosphere outside and to surrounding structures. In addition, the equivalent of 51 cfm of unfiltered outside air enters through the main control room doors and other sources. Isolation dampers located in each intake line may be selectively closed by control room personnel. The selection between the two would be based on the objective of admitting a minimum of airborne activity to the control room via the makeup airflow.

The control room ventilation flow system is shown in Figure 9.4-1.

To evaluate the ability of the control room to meet the requirements of General Design Criterion 19, a time-dependent model of the control room was developed. In this model, the outside air concentration enters the control room via the isolation damper bypass line and the HEPA filters and charcoal absorbers. The concentration in the room is reduced by decay, leakage out, and by recirculation through the HEPA filters and charcoal absorbers. Credit for filtration is taken during two passes through the charcoal absorbers. Using these assumptions, the following equations for the rate of change of the control room concentrations are obtained:

$$\frac{dM}{dt} = C_o(1-K_1)L/V - (L/V)M - \frac{R_c}{V}M - \lambda M \qquad (1)$$

$$\frac{dN}{dt} = \frac{R_c}{V}(1-K_2)\ M - (L/V)N - \lambda N \qquad (2)$$

$$C(t) = M(t) + N(t) \qquad (3)$$

Where:

M(t) = Once-filtered time-dependent concentration

N(t) = Twice-filtered (or more) time-dependent concentration

C(t) = Total time-dependent concentration in control room

C_o = Concentration of isotope entering air intake

K_1 = Filter efficiency for a particular isotope during first pass

K_2 = Filter efficiency for a particular isotope during second pass

L = Flow rate of outside air into control room and leakage out of control room

R_c = Recirculated air flow rate through filters

λ = Decay constant

V = Control room free volume

These equations are readily solvable if C_o is constant or a simple function of time during a time interval. Since C_o consists of a number of terms involving exponentials, it was assumed to be constant during particular time intervals corresponding to the average concentration during each interval as described below. Solving equations (1), (2), and (3) yields:

$$C(t) = \left[\frac{1-K_1 1-K_2 C_o}{W_m V}\right] \times \left[\frac{L}{(1-K_2)}(1-e^{-Wmt}) + \frac{R_c L}{W_n V}(1-e^{-W_n t}) - L(e^{-W_n t} - e^{-W_m t})\right] \qquad (4)$$

Where:

$$W_m = \frac{(L + R_c + \lambda V)}{V}$$

$$W_n = \frac{(L + \lambda V)}{V}$$

The value of C_o used in equation (4) is determined as follows:

$$C_{oi} = (X/Q)_i \frac{\int_{t_i}^{t_{i+1}} R\,dt}{t_{i+1} - t_i} \qquad (5)$$

C_{oi}= Average concentration of activity outside control room during ith time period (Ci/m^3).

$(x/Q)_i$= Atmospheric dilution factor (sec/m^3) during the ith time period.

R= Time dependent release rate of activity from containment (Ci/sec).

The atmospheric dilution factors were determined using the accumulated meteorological data on wind speed, direction, and duration of occurrence obtained from the Watts Bar plant site applied to a building wake dilution model. The dilution factors are calculated by the ARCON96 methodology[8] and are the maximum values for each time period. The worst case is Unit 1 exhaust to intake 2. These factors are applied for the first 8 hours, at which time it is assumed that the operator selects intake 1 which has more favorable dilution factors. The values used in the analysis are given in Table 15.5-14.

Equation (4) is used to determine the concentration at any time within a time period and upon integrating and dividing by the time interval gives the average concentration during the time interval due to inflow of radioactivity with outside air as shown:

$$\overline{C_i} = \int_0^T \frac{C_i(t)dt}{T - 0} \qquad (6)$$

Where:

$T = t - t_{i-1}$

t = Time after accident

t_{i-1} = Time at end of previous time period

Further contributions to the concentration during the time period are due to the concentrations remaining from prior time periods. These contributions are obtained from the following equations:

$$C_{R(i+j)} = M_{R(i+j)} + N_{R(i+j)} \qquad (7)$$

$$\frac{dM_{R(i+j)}}{dt} = -(L/V + R_c/V + \lambda)M_R(i+j) \qquad (8)$$

$$\frac{dN_{R(i+j)}}{dt} = (R_c/V)(1-K_2)M_{R(i+j)} - (L/V+\lambda)N_{R(i+j)} \qquad (9)$$

With initial conditions:

$M_{R(i+j)}(0) = M_{R0(i)} =$ (Once-filtered concentration at end of the ith time period.)

$N_{R(i+j)}(0) = N_{R0(i)} =$ (Twice-filtered, or more, concentration at end of the ith time period.)

Solving equations (8) and (9) and substituting certain initial condition relations, equation (7) becomes:

$$C_{R(i+j)} = C_{R0(i)}e^{-W_N(t-ti)} - M_{R0(i)}K_2(e^{-W_N(t-ti)} - e^{-W_M(t-ti)}) \qquad (10)$$

Integrating equation (10) for each of the prior time periods gives the contribution from these time periods to the present time period. The average concentration is determined for these contributions using the method of equation (6).

Filter efficiencies of 95% for elemental and particulate iodine and 95% for organic iodine were deemed appropriate for the first filter pass. Since the concentrations of iodine in the main control room are such reduced as a result of this filtration, the efficiencies were reduced for the second pass to 70% for elemental and particulate iodine, and 70% for organic iodine.

To account for the unfiltered inleakage, a bypass leak rate (BPR) of 51 cfm was added to the makeup flow (L in equation (1)) of 711 cfm, and the filter factor for the first pass was decreased by the ratio L/(L+BPR). The filter efficiencies for the second pass are not affected by the unfiltered inleakage.

The filter efficiency for noble gases was taken as zero for all cases.

The above equations were incorporated into computer program COROD[17] together with appropriate equations for computing gamma dose, beta dose, and inhalation dose using these average nuclide concentrations and time periods. The whole body gamma dose calculation consists of an incremental volume summation of a point kernel over the control room volume. The principal gammas of each isotope are used to compute the dose from each isotope. The dose computations for beta activity were based on a semi infinite cloud model. Doses to thyroid were based on activity to dose conversion factors. (The equations and various data are given below.) The doses from these calculations are presented in Table 15.5-9. Gamma dose contributions from shine through the control room roof due to the external cloud and from shine through the control room walls from adjacent structures and from containment are computed using an incremental volume summation of a point kernel which includes buildup factors for the concrete shielding. For the calculation of shine through the control room roof, an atmospheric, rectangular volume several thousand feet in height and several control room widths was used. The control room roof is a 2 foot 3-inch-thick concrete slab and is the only shielding considered in this calculation. The average isotope concentrations at the control bay for each time period were used as the source concentrations. For the shine from adjacent structures, the shielding consists of the 3-foot-thick (5 feet in certain areas) control room walls. The doses are calculated similarly to the shine dose through the roof. The average isotope concentrations at the control bay intake for each time period are also used for these calculations.

The shine from the spreading room below the control room is also computed in the same manner as adjacent structures.

Shielding for this computation consists of the 8-inch-thick concrete floor. The summation of the incremental elements is performed over the volume of each room or structure of interest.

In addition to the dose due to shine from surrounding structures and from the passing cloud, the shine from the reactor containment building also contributes to the gamma whole body dose to personnel. This contribution is computed in the same manner as the methods used above. Due to the location of the Auxiliary Building between the Reactor Buildings and the control room and the thicker control room auxiliary building wall near the roof, the minimum ray path through concrete from the containment into the control room below 10 feet above the control floor, is 8 feet. All nuclides released to containment are assumed uniformly distributed and their time-dependent concentrations were used to compute the dose. The dose computed from this source is small.

Several doors penetrate the control room walls, and the dose at these areas would be larger than the doses calculated as described above. The potential shine at these doors and at other penetrations has been evaluated. As a result, hollow steel doors filled with no. 12 lead shot have been incorporated into the design of the shield wall between the control room and the Turbine Building. These doors provide shielding comparable to the concrete walls. Shine through other penetrations was found to be negligible.

Another contribution to the total exposure of control room personnel is the exposure incurred during ingress from and egress to the exclusion area boundary. The doses due to ingress and egress were computed based on the following assumptions:

(1) Five minutes are required to leave the control room and arrive at car or vice versa.

(2) The distance traveled on the access road to the site exclusion boundary is estimated to be 1500 meters. The average car speed is assumed to be 25 mph.

(3) One one-way trip first day, one round-trip/day 2nd through 30th days.

The control room occupancy factors used in this calculation were taken from Murphy and Campe[9]. They are:

100%	occupancy 0-24 hours
60%	occupancy 1-4 days
40%	occupancy 4-30 days.

All atmospheric dilution factors were conservatively based on 5th percentile wind velocity averages.

It was also assumed that initially the makeup air intake would be through the vent admitting the highest radioisotope concentration, but that the main control room personnel would switch intake vents 8 hours after the accident in order to admit a lower amount of airborne activity to the MCR via the makeup air flow.

The whole body, beta, and thyroid doses from the radiation sources discussed above are presented in Table 15.5-9. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose Equations, Data, and Assumptions

The dose from gamma radiation originating within the control room is given by:

$$D_\Upsilon = 1.69 \times 10^4 \sum_{i=1}^{\alpha}\left[\sum_{k=1}^{\beta} TCOT_{ik}\left(\sum_{l=1}^{\Upsilon}\left\{E_{kl}f_{kl}\left(\frac{\mu_e}{\rho}\right)_l \sum_{m=1}^{\varepsilon}\sum_{n=1}^{\omega}\sum_{q=1}^{\sigma}\frac{\exp(-\mu_{al}\sqrt{x_m^2+y_n^2+z_q^2})}{(x_m^2+y_n^2+z_q^2)}\cdot \Delta x\Delta y\Delta z\right\}\right)\right] \quad (11)$$

Where:

$D\gamma$ = Absorbed dose in flesh in mrads

$TCOT_{ik}$=Total concentration integrated over time period i of isotope k in curies/m^3

$E_{k\ell}$ = Energy of gamma ℓ from isotope k in MeV

$f_{k\ell}$ = Number of ℓ gammas of isotope k given off per disintegration

$\left(\frac{\mu_e}{\rho}\right)_\ell$ = Mass attenuation coefficient for flesh determined at the energy of gamma ℓ in cm^2/gram

$\mu_{\alpha\tau}$=Linear attenuation coefficient for air determined at the energy of gamma ℓ in inverse meters

x_m, y_n, z_q=Coordinate distances from the dose point to the source volume element (m,n,q) in meters

$\Delta x, \Delta y, \Delta z$=Dimensions of source element (m,n,q)

α=Number of time periods

β=Number of isotopes

γ=Number of gammas from an isotope

ε=Number of intervals in the x direction

ω=Number of intervals in the y direction

σ=Number of intervals in the z direction

The control room radiation dose from gamma radiation originating outside of the control room and penetrating concrete walls is given as:

$$D_\Upsilon = 1.69 \times 10^4 \sum_{i=1}^{\alpha} \left[\sum_{k=1}^{\beta} C_{o_{ik}} \left(\sum_{l=1}^{\Upsilon} \left\{ E_{kl} f_{kl} \left(\frac{\mu_e}{\rho}\right)_l \sum_{m=1}^{\varepsilon} \sum_{n=1}^{\omega} \sum_{q=1}^{\sigma} \frac{\exp(-\mu_{al}\sqrt{x_m^2 + y_n^2 + z_q^2})}{(x_m^2 + y_n^2 + z_q^2)} \cdot \exp(-\mu_{cl} t_c \sec\theta) \right.\right.\right.$$
$$\left.\left.\left. \cdot\ B_c(\mu_{cl} t_c \sec\theta) \cdot \Delta x \Delta y \Delta z \right\} \right) \right] (t_i - t_{i-1})$$

Where:

μ_{cl} = Linear attenuation coefficient of concrete determined at the energy of gamma ζ in inverse meters

t_c= Concrete shield thickness in meters

θ = Angle between a vector normal to the shield and a vector from the dose point to the source point

$B_c(\mu_{cl}t_c\sec\theta)$ = Buildup factor for concrete

C_{oik} = Average concentration of isotope k outside the control room during time period i in curies/m^3

t_{i-1},t_i = Times at the beginning and end of time period i in hours

Other parameters are defined as previously noted.

The dose from beta radiation is given by the semi-infinite cloud immersion dose:

$$D_B = (0.230)\,(X/Q)\left[\sum_{i=1} Q \sum_{k=1} E_{ik}f_{ik}\right] \qquad (12)$$

Where:

D_B=Dose due to beta in rem

X/Q=Atmospheric dispersion factor during time period in sec/m^3

Q_i=Accumulated activity release of isotope i during time period

E_{ik}=Average energy of beta k of isotope i

f_{ik}=Number of k betas of isotope i per disintegration

For beta dose in the control room, equation (12) becomes:

$$D_B = (0.230)\sum_{i=1}^{\delta}\sum_{i=1}^{\alpha}\bar{C}_{ij}\sum_{1}^{\beta}E_{ik}f_{ik}(tj - t_{j-1})$$

Where:

$\overline{C}_{ij}$ =Average concentration of isotope i during time period j

Inhalation Dose (Thyroid)

The inhalation dose for a given period of time has the general form:

$$D_I = (X/Q)(B)\left[\sum_{i=1}^{n}(Q_{i_j})(DCF_i)\right](t_J - t_{j-1}) \qquad (13)$$

Where:

D_I=Thyroid inhalation dose, rem

X/Q=Site dispersion factor during time period, sec/m^3

B=Breathing rate during time period, m^3/hr

Q_{ij}=Average activity release rate during time period j of iodine isotope i

DCF_i=ICRP-30 Dose conversion factor for iodine isotope i, rem/microcurie inhaled

t_j=Total time at end of period j, hours

For inhalation dose within the control room, equation (13) becomes:

$$D_I = (B)\left[\sum_{i=1}^{n} C_{ij}(DCF_i)\right](t_j - t_{j-1})$$

In this expression C_{ij}, the average concentration of isotope i during time period j, has replaced the following factor:

(X/Q) Q_{ij}

The C_{ij}'s are those determined by equations (4) and (6). The breathing rate factor B, was taken to be 3.47 x 10^{-4} m^3/sec, 1.75 x 10^{-4} m^3/sec, and 2.32 x 10^{-4} m^3/sec for the time intervals of 0-8 hours, 8-24 hours, and 24 hours - 30 days, respectively.

15.5.4 Environmental Consequences of a Postulated Main Steam Line Break

The postulated accidents involving release of steam from the secondary system will not result in a release of radioactivity unless there is a leakage from the reactor coolant system to the secondary system in the steam generator. An acceptable primary-to-secondary leakage rate for the main steam line break (MSLB) accident is 1 gallon per minute (gpm) for the faulted steam generator loop and 150 gallons per day (gpd) for each unfaulted steam generator.

A calculation determines the offsite and main control room doses resulting from a MSLB incorporating the above primary-to-secondary criteria. The calculation determined that 1 gpm (at standard temperature and pressure) primary-to-secondary leakage in the faulted steam generator results in site boundary doses within 10CFR100 guidelines and control room doses within the 10CFR50, Appendix A, General Design Criteria (GDC)-19 limit. The calculation uses TVA computer codes STP, FENCDOSE and COROD. The STP output is input to COROD, which determines control room operator dose and FENCDOSE, which determines the 30-day low population zone (LPZ) and the 2-hour exclusion area boundary (EAB) dose.

Two methods for determining the resultant dose for a MSLB in accordance with the Standard Review Plan 15.1.5, Appendix A methodology are:

1. A pre-accident iodine spike where the iodine level in the reactor coolant spiked upward to the maximum allowable limit of 14 μCi/gm I-131 dose equivalent just prior to the initiation of the accident.

2. The reactor coolant at the maximum steady state dose of equivalent I-131 of 0.265 μCi/gm with an accident initiated iodine spike consisting of a 500 times increase on the rate of iodine release from the fuel.

In both cases, the primary-to-secondary side leak is assumed to be 1 gpm in the faulted steam generator loop and 150 gpd in each faulted loop. The primary side activity release uses the Technical Specification (TS) limit design reactor coolant activities, and the secondary side activity uses the Technical Specification limit of $\leq$0.1μCi/gm I-131 dose equivalent.

The steam releases to the atmosphere for the MSLB are in Table 15.5-16.

The gamma, beta and thyroid doses for the MSLB accident at the EAB and LPZ are in Table 15.5-17. The doses from this accident are less than the reference values as listed in 10CFR100 (25 rem whole body and 300 rem thyroid).

The whole body, beta and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-17. The doses are calculated by the COROD computer code. [17] Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the GDC limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] determine the dose. Dose conversion factor in ICRP-30 [25] determine thyroid doses in place of those found in TID-14844.

Assumptions for the MSLB accident:

1. RCS letdown flow of 124.39 gpm.
2. RCS letdown demineralizer efficiency is 1.0 for iodines.
3. ANSI/ASN-18.1-1984 spectrum scaled up to 0.265 or 14 μCi/gm equivalent iodine.
4. Two cases were used. In the first, pre-accident iodine spike of 14 μCi/gm I-131 dose equivalent in the RCS. In the second case, an accident initiated spike which increases the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.
5. Primary side to secondary side leakage of 150 gpd (standard temperature and pressure) per steam generator in the intact loops.
6. The primary-to-secondary leakage mass release to the Environment is 1 gpm (standard temperature and pressure) from the faulted loops.
7. Steam generator secondary inventory released as steam to the atmosphere:

   a) total from the non-defective steam generators (0-2 hrs), 433,079 lbm

   b) total from the non-defective steam generators (2-8 hrs), 870,754 lbm

   c) total from the faulted steam generator (0-30 mins), 96,100 lbm
8. Iodine partition coefficients from steaming of steam generator water:

   i. non-defective steam generators initial inventory and primary-to-secondary leakage, 100.

   ii. faulted steam generator initial inventory and primary-to-secondary leakage, 1.0
9. Atmospheric dilution factors, x/Q, are in Table 15A-2 for offsite and Table 15.5-14 for control room personnel.
10. Main control room related assumptions are in Table 15.5-14.

15.5.5 Environmental Consequences of a Postulated Steam Generator Tube Rupture

Thermal and hydraulic analyses determine the plant response for a design basis steam generator tube rupture (SGTR), and the integrated primary to secondary break flow and mass releases from the ruptured and intact steam generators (SGs) to the condenser and the atmosphere (Section 15.4.3). An analysis of the environmental consequences of the postulated SGTR has also been performed, utilizing the reactor coolant mass and secondary steam mass releases determined in the base thermal and hydraulic analysis (See Reference [38] in Section 15.4). Table 15.5-18 summarizes the parameters used in the SGTR analysis.

The SGTR thermal and hydraulic analysis documents use WBN specific parameters and actual operator performance data, as determined from simulator exercises utilizing the appropriate emergency operating procedures (EOPs). Two cases were analyzed. Case 1: The primary side activity release use the maximum Technical Specification (TS) limit design reactor coolant activities and an iodine spike immediately after the accident that increases the iodine activity in the reactor coolant by a factor of 500 times the iodine production rate necessary to maintain a steady state concentration of 0.265 µCi/gm of I-131 dose equivalent. Case 2: The initial reactor coolant activity is at 14 µCi/gm of I-131 dose equivalent due to a pre-accident iodine spike caused by an RCS transient. For both cases, the secondary side releases uses expected secondary side activities, based on ANSI/ANS-18.1-l984[14] as modified for WBN, and on a 150 gpd/steam generator primary-to-secondary-side leakage. Credit was taken for flashing of the primary coolant (References [34] and [35] of Section 15.4), but "scrubbing" of the iodine in the rising steam bubbles by the water in the steam generator was conservatively neglected. A partition factor of 100 applies to iodine in the remaining unflashed coolant which will boil.

The atmospheric diffusion coefficients (X/Q) for the exclusion area boundary (EAB) and offsite dose determination are the same as those used for the LOCA analysis (Appendix 15A). The X/Q values for the control room operator were determined in the analysis. The LOCA X/Q values are for a release from the shield building vent, whereas the SGTR release is from the top of the main steam valve vault. The methodology for determination of the WBN control room X/Q values is based on computer code ARCON96.

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are in Table 15.5-19. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are in Table 15.5-14. The dose to whole body is below the GDC 19 limit of 5 rem for control room personnel, and thyroid dose is below the limit of 30 rem.

Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The gamma, beta, and thyroid dose for the SGTR event are in Table 15.5-19. The doses at the EAB and the low population zone are less than 10% of the 10 CFR 100 limits.

15.5.6 Environmental Consequences of a Postulated Fuel Handling Accident

The analysis of the fuel handling accident considers three cases. The first case is for a Fuel Handling Accident inside containment with the containment closed and the Reactor Building Purge System operating. This analysis is discussed in Section 15.5.6.1 and is based on Regulatory Guide 1.25[11] and NUREG 5009[24]. The second case is for an accident in the spent fuel pool area located in the Auxiliary

Building. This case is discussed in Section 15.5.6.2 and is evaluated using the Alternate Source Term based on Regulatory Guide 1.183[18], "Alternate Source Terms." The third case considered is an open containment case for an accident inside containment where there is open communication between the containment and the Auxiliary Building. This evaluation is discussed in Section 15.5.6.2 and is based on Regulatory Guide 1.183.

15.5.6.1 Fuel Handling Accident Based on Regulatory Guide 1.25

The parameters used for this analysis are listed in Table 15.5-20.

The bases for the Regulatory Guide 1.25 evaluation are:

(1) In the Regulatory Guide 1.25 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.25 analysis damage is assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) For the Regulatory Guide 1.25 analysis all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 10% of the total noble gases and radioactive iodine inventory in the rods at the time of the accident with the following gap percentage exceptions which are based on NUREG/CR 5009 [24] as appropriate: 14% of the Kr-85, 5% of the Xe-133, 2% of the Xe-135, and 12% of the I-131.

(5) Noble gases released in the containment are released through the Shield Building vent to the environment.

(6) In the Regulatory Guide 1.25 analysis the iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

(7) A filter efficiency of 90% for inorganic iodine and 30% for organic iodine for the purge air exhaust filters is used since no relative humidity control is provided.

(8) No credit is taken for natural decay after the activity has been released to the atmosphere.

(9) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.2 Fuel Handling Accident Based on Regulatory Guide 1.183

The analysis of a postulated fuel handling accident in the Auxiliary Building refueling Area is based on Regulatory Guide 1.183. i.e., Alternate Source Terms (AST). The parameters used for this analysis are listed in Table 15.5-20.a.

The bases for evaluation are:

(1) In the Regulatory Guide 1.183 analysis, the accident occurs 100 hours after plant shutdown. Radioactive decay of the fission product inventory during the interval between shutdown and placement of the first spent fuel assembly into the spent fuel pit is taken into account.

(2) In the Regulatory Guide 1.183 analysis, damage was assumed for all rods in one assembly.

(3) The assembly damaged is the highest powered assembly in the core region to be discharged. The values for individual fission product inventories in the damaged assembly are calculated assuming full-power operation at the end of core life immediately preceding shutdown. Nuclear core characteristics used in the analysis are given in Table 15.5-21. A radial peaking factor of 1.65 is used.

(4) The Regulatory Guide 1.183 analysis assumes all of the gap activity in the damaged rods is released to the spent fuel pool and consists of 8% I-131, 10% Kr-85, and 5% of other noble gases and other halogens.

(5) Noble gases released to the Auxiliary Building spent fuel pool are released through the Auxiliary Building vent to the environment.

(6) In the Regulatory Guide 1.183 analysis, the iodine gap inventory is composed of inorganic species (99.85%) and organic species (0.15%).

(7) In the Regulatory Guide 1.183 analysis, the overall inorganic and organic iodine spent fuel pool decontamination factor is 200.

(8) In the Regulatory Guide 1.183 analysis, all iodine escaping from the Auxiliary Building spent fuel pool is exhausted unfiltered through the Auxiliary Building vent.

(9) No credit is taken for the ABGTS or Containment Purge System Filters in the analysis.

(10) No credit is taken for natural decay either due to holdup in the Auxiliary Building or after the activity has been released to the atmosphere.

(11) The short-term (i.e., 0-2 hour) atmospheric dilution factors at the exclusion area boundary and low population zone given in Table 15A-2 are used. The thyroid dose utilizes ICRP-30 [25] iodine dose conversion factors. Doses are based on the dose models presented in Appendix 15A.

15.5.6.3 Fuel Handling Accident Results

The radiation dose results of the Regulatory Guide 1.25 with the containment closed fuel handling accident (FHA) are given in Table 15.5-23. For a FHA inside containment, no allowance has been made for possible holdup or mixing in the primary containment or isolation of the primary containment as a result of a high radiation signal from the monitors in the ventilation systems for the case where containment penetrations are closed to the Auxiliary Building. However, the containment purge filters are credited. Dose equations in TID-14844 [23] were used to determine the dose. Dose conversion factors in ICRP-30 [25] were used to determine thyroid doses in place of those found in TID-14844.

The ventilation function of the reactor building purge ventilating system (RBPVS) is not a safety-related function. However, the filtration units and associated exhaust ductwork do provide a safety-related filtration path following a fuel-handling accident prior to automatic closure of the associated isolation valves. The RBPVS contains air cleanup units with prefilters, HEPA filters, and 2-inch-thick charcoal adsorbers. This system is similar to the auxiliary building gas treatment system except that the latter is equipped with 4-inch-thick charcoal adsorbers. Anytime fuel handling operations are being carried on inside the primary containment, either the containment is isolated or the reactor building purge filtration system is operational. The assumptions listed above are, therefore, applicable to a fuel handling accident inside primary containment.

The thyroid, gamma, and beta doses for FHAs for the closed containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10CFR100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body. These doses are calculated by using Revision 5 of the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel and the thyroid dose is below the limit of 30 rem.

The radiation dose results of the Regulatory Guide 1.183 fuel handling accident (FHA) are given in Table 15.5-23. Alternate source term (AST) described in RG 1.183 was selectively used to evaluate the FHA due to an event in the spent fuel pool located in the Auxiliary Building or in the containment when the equipment hatch or both doors in a personnel air lock are open. As part of this selective implementation of AST, the following assumptions are used in the analysis:

- The total effective dose equivalent (TEDE) acceptance criterion of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11.
- The gap activity is revised to be consistent with that required by RG 1.183.
- The decontamination factors were changed to be consistent with those required by RG. 1.183.
- No Auxiliary Building isolation is assumed.
- No filtration of the release from the spent fuel pool to the environment by the ABGTS is assumed.

The evaluation for the FHA at the spent fuel pool is a bounding analysis for a dropped assembly in containment when the containment is open. The release point for the containment purge system is the Unit 2 shield building stack. The X/Qs are lower for this release point than for the normal auxiliary building exhaust. In addition, any release from the shield building stack would go through the purge system HEPA and Charcoal filter assemblies prior to release. Currently, when the purge lines isolate on high radiation, the auxiliary building also isolates and ABGTS is actuated. The release point for ABGTS is the shield building stacks and the releases are filtered through HEPA and Charcoal assemblies. Thus AST analysis for the FHA in the Auxiliary Building that considers no filtration is conservative and an acceptable as the basis for the containment open evaluation.

The thyroid, gamma, and beta doses for FHAs in the Auxiliary and the open containment are given in Table 15.5-23 for the exclusion area boundary and low population zone. These doses are less than 25% of the 10 CFR 100.11 limits of 300 rem to the thyroid, and 25 rem gamma to the whole body and less than the 10 CFR 50.67 limit of 25 rem TEDE. These doses are calculated by using Revision 5 of the computer code FENCDOSE [16].

The whole body, beta, and thyroid doses to control room personnel from the radiation sources discussed above are presented in Table 15.5-23. The doses are calculated by the COROD computer code [17]. Parameters for the control room analysis are found in Table 15.5-14. The dose to whole body is below the 10 CFR 50 Appendix A, GDC 19 limit of 5 rem for control room personnel, and the thyroid dose is below the limit of 30 rem and the 10 CFR 50.67 limit of 5 rem TEDE.

15.5.7 Environmental Consequences of a Postulated Rod Ejection Accident

This accident is bounded by the loss-of-coolant accident. See Section 15.5.3 for the loss-of-coolant accident.

REFERENCES

(1) Styrikovich, M. A., Martynova, 0. I., Katkovska, K. YA., Dubrovski, I. YA., Smrinova, I. N., "Transfer of Iodine from Aqueous Solutions to Saturated Vapor," translated from Atomnaya Energiya, Vol. 17, No. 1, pp. 45-49, July 1964.

(2) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.24, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

(3) Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, June 1974.

(4) D. D. Malinowski, "Iodine Removal in the Ice Condenser System," WCAP-7426, April 1970.

(5) NAA-SR 10100, Conventional Buildings for Reactor Containment.

(6) ORNL-NSIC-4, Behavior of Iodine in Reactor Containment Systems, February 1965.

(7) Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident."

(8) Ramsdell, J. V. Jr. and C. A. Simonen, "Atmospheric Relative Concentrations in Building Wakes." Prepared by Pacific Northwest laboratory fo the U. S. Nuclear Regulatory Commission, PNL-10521, NUREG/CR-6331, Revision1, May 1997.

(9) K. G. Murphy and Dr. K. M. Campe "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

(10) Deleted by Amendment 80.

(11) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Division of Reactor Standards, U.S. Atomic Energy Commission, March 23, 1972.

(12) Regulatory Guides for Water Cooled Nuclear Power Plants, Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Directorate of Regulatory Standards, U.S. Atomic Energy Commission, May 1974.

(13) D. B. Risher, Jr., "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," WCAP-7588, Revision 1, December 1971.

(14) ANSI/ANS-18.1-1984, "Radioactive Source Terms for Normal Operations of Light Water Reactors," December 31, 1984.

(15) WCAP-7664, Revision 1, "Radiation Analysis Design Manual-4 Loop Plant," RIMS Number NEB 810126 316, October 1972.

(16) Computer Code FENCDOSE, Code I.D. 262358.

(17) Computer Code COROD, Code I.D. 262347.

(18) Regulatory Guide 1.183 R0, Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, US Nuclear Regulatory Commission, July 2000.

(19) Not used

(20) NRC Safety Evaluation for Watts Bar Nuclear Plant Unit 1, Amendment 38, for Steam Generator Tubing Voltage Based Alternate Repair Criteria for Outside Diameter Stress Corrosion Cracking (ODSCC) dated February 26, 2002.

(21) NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking", dated August 3, 1995.

(22) TVA Letters to NRC "Technical Specification Change No. WBN-TS-99-014 - Steam Generator Alternate Repair Criteria for Axial Outside Diameter Stress Corrosion Cracking (ODSCC)," dated April 10, 2000, September 18, 2000, August 22, 2001, November 8, 2001 and January 15, 2002.

(23) J.J. Dinunno, et, al "Calculation of Distance Factors for Power and Test Reactor Sites", TIC-14844, March 1962.

(24) NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," February 1988.

(25) International Commission on Radiation Protection (ICRP) Publication 30, Limits for Intakes of Radionuclides by Workers," 1979.

Table 15.5-1 Parameters Used In Loss Of A. C. Power Analyses

	Realistic Analysis	Conservative Analysis
Core thermal power	3565 MWt	3565 MWt
Steam generator tube leak rate prior to and during accident	1 gpm	1.0 gpm
Fuel defects	ANSI/ANS 18.1 - 1984	Technical Specification limit of 0.1 µCi/gm I-131 dose equivalent
Iodine partition factor in steam generator prior to and during accident	0.01	0.01
Blowdown rate per steam generator prior to accident	25 gpm	25 gpm
Duration of plant cooldown by secondary system after accident	8 hrs	8 hrs
Steam release from 4 steam generators	444,875 lbm (0-2 hrs) 903,530 lbm (2-8 hrs)	444,875 lbs (0-2 hrs) 903,530 lbs (2-8 hrs)
Meteorology	See Tables 15A-2 & 15.5-14	See Tables 15A-2 & 15.5-14

Table 15.5-2 Doses From Loss Of A/C Power

Conservative Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	7.45E-04	4.18E-04	2.10E-04
Beta	4.48E-04	2.52E-04	2.52E-03
Thyroid - ICRP-30	4.57E-02	2.57E-02	2.09E-02

Realistic Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	1.80E-08	1.01E-08	5.05E-09
Beta	1.66E-05	9.29E-06	1.79E-04
Thyroid - ICRP-30	1.10E-06	6.18E-07	5.03E-07

Table 15.5-3 Parameters Used In Waste Gas Decay Tank Rupture Analyses

	Realistic Analysis	**Regulatory Guide 1.24 Analysis**
Core thermal power	3565 MWt	3565 MWt
Plant load factor	1.0	1.0
Fuel defects	ANSI/ANS-18.1, 1984	1%
Activity released from GWPS	(1)	See Table 15.5-4
Time of accident	After Tank Fill	At end of equilibrium core cycle
Meteorology	See Table 15.5-14 and Table 15A-2	See Table 15.5-14 and Table 15A-2

(1)Activity based on maximum concentrations of each isotope and actual plant flow rates of the GWPS.

Table 15.5-4 Waste Gas Decay Tank Inventory (One Unit) (Regulatory Guide 1.24 Analysis)

Isotope	Activity (Curies)
Xe-131m	8.9×10^{2}
Xe-133	6.8×10^{4}
Xe-133m	1.0×10^{3}
Xe-135	9.4×10^{2}
Xe-135m	4.8×10^{1}
Xe-137	2.7×10^{-1}
Xe-138	3.2
Kr-83m	1.7×10^{1}
Kr-85	4.2×10^{3}
Kr-85m	1.3×10^{2}
Kr-87	2.9×10^{1}
Kr-88	1.6×10^{2}
Kr-89	1.0×10^{-1}
I-131	4.8×10^{-2}
I-132	---------
I-133	3.3×10^{-2}
I-134	---------
I-135	1.2×10^{-2}

Table 15.5-5 Doses From Gas Decay Tank Rupture

Regulatory Guide 1.24 Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	5.96E-01	1.67E-01	8.43E-01
Beta	1.61E+00	4.51E-01	7.28E+00
Thyroid - ICRP-30	1.29E-02	3.60E-03	6.99E-03

Realistic Analysis (rem)	2HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	2.88E-02	8.05E-03	3.81E-02
Beta	1.10E-01	3.08E-02	5.01E-01
Thyroid - ICRP-30	1.21E-02	3.37E-03	6.50E-03

Table 15.5-6 Parameters Used In LOCA Analysis (Page 1 of 2)

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Primary containment free volume	1.27×10^6 ft^3
Annulus free volume	3.75×10^5 ft^3
Primary containment deck (air return) fan flow rate	40,000 cfm
Number of deck (containment air return fans) fans assumed operating	1 of 2
Activity released to primary containment and available for release	
noble gases	100% of core inventory
iodines	25% of core inventory
Form of iodine activity in primary containment available for release	
elemental iodine	91%
methyl iodine	4%
particulate iodine	5%
Ice condenser removal efficiency for elemental and particulate iodine	See Table 15.5-7
Primary containment leak rate (volume percent)	0.25% per day (0-24 hours) 0.125% per day (1-30 days)
Percent of primary containment leakage to auxiliary building	25%
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Delay time of activity in auxiliary building before ABGTS operation	None
Delay time before filtration credit is taken for the ABGTS	4 min
Mean holdup time in auxiliary building after initial 4 minutes	0.3 hours
ABGTS flow rate	9000 cfm

Table 15.5-6 Parameters Used In LOCA Analysis (Page 2 of 2)

Leakage from Auxiliary Building to ABGTS downstream HVAC (bypass of filters)	27.88 cfm
Leakage from ABGTS HVAC into Auxiliary Building	8.87 cfm
Leakage from Auxiliary Building into EGTS downstream HVAC (bypass of filters)	10.7 cfm
Leakage from Auxiliary Building to environment due to single failure of ABGTS (from 30 minutes to 34 minutes post-LOCA)	9900 cfm (for 4 minutes)
Percent of primary containment leakage to annulus	75%
Emergency gas treatment system flow rates	See Table 15.5-8 and Table 15.5-8A
Percent of annulus free volume available for mixing of recirculated activity	50%
Number of emergency gas treatment system air handling units operating	1 of 2
Emergency gas treatment system filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Shield building mixing model (see Section 15.5.3)	50% mixing
Meteorology	See Table 15.5-14 and Table 15A-2

Table 15.5-7 Ice Condenser Elemental And Particulate Iodine Removal Efficiency[1,2]

Time Interval Post LOCA (Hours)	Iodine Removal Efficiency
0.0 to 0.156	0.96
0.156 to 0.267	0.76
0.267 to 0.323	0.73
0.323 to 0.489	0.71
0.489 to 0.615	0.60
0.615 to 0.768	0.58
0.768 to 0.824	0.40
0.824 to 720	0.0

(1) The ice condenser removal efficiencies given in the above table are used for the Regulatory Guide 1.4 analysis. The inlet steam/air mixture coming into the ice condenser is greater than 90% steam by volume initially due to the delaying of the operation of the containment deck fans. Without the delay of operation of the deck fans, the amount of steam by volume in the inlet mixture initially would be much lower and the ice condenser iodine removal efficiencies would be reduced.

(2) The ice bed iodine removal efficiency, O_I, has been computed on a time dependent basis and is shown in Table 15.5-7. Note that the information presented in Table 15.5-7 has been revised by Westinghouse letter WAT-D-10954. The revised efficiency information is associated with the WCAP-15699, Revision 1 analysis for reduced ice weight. A comparison of the information presented in Table 15.5-7 and the revised information contained in WAT-D-10954 shows that the information in Table 15.5-7 is conservative. Analyses supporting the plant design basis acknowledge the revised efficiency information but shall utilize the information presented in Table.15.5-7.

Table 15.5-8 EMERGENCY GAS TREATMENT SYSTEM FLOW RATES (Sheet 1 of 1)

Time Interval (sec)	Time Interval (hours)	Recirculation Rate (cfm)	Recirculation Rate (cfh)	Exhaust Rate (cfm)	Exhaust Rate (cfh)
0-30	0-0.0083	0.00	0.00E+00	0.00	0.00E+00
30-39	0.0083-0.0108	3600.00	2.16E+05	0.00	0.00E+00
39-40	0.0108-0.0111	3286.62	1.97E+05	313.38	1.88E+04
40-41	0.0111-0.0114	2352.31	1.41E+05	1247.69	7.49E+04
41-42	0.0114-0.0117	1304.79	7.83E+04	2295.21	1.38E+05
42-43	0.0117-0.0119	362.60	2.18E+04	3237.40	1.94E+05
43-190	0.0119-0.0528	0.00	0.00E+00	3600.00	2.16E+05
190-191	0.0528-0.0531	537.28	3.22E+04	3062.72	1.84E+05
191-192	0.0531-0.0533	733.23	4.40E+04	2866.77	1.72E+05
192-193	0.0533-0.0536	735.14	4.41E+04	2864.86	1.72E+05
193-194	0.0536-0.0539	737.51	4.43E+04	2862.49	1.72E+05
194-199	0.0539-0.0553	745.23	4.47E+04	2854.77	1.71E+05
199-207	0.0553-0.0575	764.12	4.58E+04	2835.89	1.70E+05
207-215	0.0575-0.0597	790.80	4.74E+04	2809.20	1.69E+05
215-225	0.0597-0.0625	825.45	4.95E+04	2774.56	1.66E+05
225-245	0.0625-0.0681	892.72	5.36E+04	2707.29	1.62E+05
245-265	0.0681-0.0736	992.80	5.96E+04	2607.20	1.56E+05
265-285	0.0736-0.0792	1102.40	6.61E+04	2497.61	1.50E+05
285-305	0.0792-0.0847	1217.05	7.30E+04	2382.95	1.43E+05
305-446	0.0847-0.1239	1664.05	9.98E+04	1935.96	1.16E+05
446-601	0.1239-0.1669	2356.72	1.41E+05	1243.29	7.46E+04
601-602	0.1669-0.1672	2661.35	1.60E+05	938.65	5.63E+04
602-1700	0.1672-0.4722	3600.00	2.16E+05	0.00	0.00E+00
1700-1701	0.4722-0.4725	3508.13	2.10E+05	91.87	5.51E+03
1701-1702	0.4725-0.4728	3423.44	2.05E+05	176.56	1.06E+04
1702-1703	0.4728-0.4731	3410.73	2.05E+05	189.27	1.14E+04
1703-1704	0.4731-0.4733	3408.66	2.05E+05	191.34	1.15E+04
1704-1705	0.4733-0.4736	3408.17	2.04E+05	191.83	1.15E+04
1705-1706	0.4736-0.4739	3407.91	2.04E+05	192.09	1.15E+04
1706-1855	0.4739-0.5153	3395.23	2.04E+05	204.77	1.23E+04
1855-2100	0.5153-0.5833	3372.37	2.02E+05	227.64	1.37E+04
2100-30 days*	0.5833-720	3350.00	2.01E+05	250.00	1.50E+04

*Required to maintain annulus pressure when assuming 250 cfm annulus inleakage

Table 15.5-8A Emergency Gas Treatment System Flow Rates (Unit 2)

Time Interval		Time Interval		Recirculation Rate		Exhaust Rate	
(sec)	(sec)	(hours)	(hours)	(cfm)	(cfh)	(cfm)	(cfh)
0	30	0	0.0083	0	0.00E+00	0	0.00E+00
30	39	0.0083	0.0108	7200	4.32E+05	0	0.00E+00
39	40	0.0108	0.0111	6573.24	3.94E+05	626.76	3.76E+04
40	41	0.0111	0.0114	4704.62	2.82E+05	2495.38	1.50E+05
41	42	0.0114	0.0117	2609.58	1.57E+05	4590.42	2.75E+05
42	43	0.0117	0.0119	725.2	4.35E+04	6474.8	3.88E+05
43	71	0.0119	0.0197	0	0.00E+00	7200	4.32E+05
71	78	0.0197	0.0217	0	0.00E+00	7200	4.32E+05
78	79	0.0217	0.0219	1062	6.37E+04	6138	3.68E+05
79	80	0.0219	0.0222	4775	2.87E+05	2425	1.46E+05
80	102	0.0222	0.0283	4337	2.60E+05	2863	1.72E+05
102	132	0.0283	0.0367	4188	2.51E+05	3012	1.81E+05
132	165	0.0367	0.0458	3922	2.35E+05	3278	1.97E+05
165	170	0.0458	0.0472	3762	2.26E+05	3438	2.06E+05
170	210	0.0472	0.0583	3719	2.23E+05	3481	2.09E+05
210	307	0.0583	0.0853	3760	2.26E+05	3440	2.06E+05
307	498	0.0853	0.1383	4050	2.43E+05	3150	1.89E+05
498	602	0.1383	0.1672	4797	2.88E+05	2403	1.44E+05
602	603	0.1672	0.1675	5232	3.14E+05	1968	1.18E+05
603	850	0.1675	0.2361	5137	3.08E+05	1432	8.59E+04
850	1100	0.2361	0.3056	5237	3.14E+05	1332	7.99E+04
1100	1350	0.3056	0.3750	5337	3.20E+05	1232	7.39E+04
1350	1600	0.3750	0.4444	5437	3.26E+05	1132	6.79E+04
1600	1850	0.4444	0.5139	5537	3.32E+05	1032	6.19E+04
1850	2100	0.5139	0.5833	5637	3.38E+05	932	5.59E+04
2100	3600*	0.5833	1.0000	5737	3.44E+05	832	4.99E+04
3600*	30 days	1.0000	30 days	3455	2.07E+05	604	3.62E+04

*Reflects operator action to place one EGTS fan in standby mode at 1 hour.

Table 15.5-9 DOSES FROM LOSS-OF-COOLANT ACCIDENT

(rem)	2Hr EAB	30 Day LPZ	Control Room
Gamma	2.12	2.18	1.05
Beta	1.25	2.61	9.10
Thyroid - ICRP - 30	40.4	14.33	3.75

Breakdown of Control Room Personnel Dose

(rem)	Airborne	Shine	Ingress/Egress	Total
Gamma	1.02	0.005	0.027	1.05
Beta	9.04	0.000	0.060	9.10
Thyroid - ICRP - 30	3.66	0.000	0.090	3.75

Table 15.5-10 Deleted by Amendment 80

Table 15.5-11 Deleted by Amendment 80

Table 15.5-12 PARAMETERS USED IN ANALYSIS OF RECIRCULATION LOOP LEAKAGE FOLLOWING A LOCA

	Regulatory Guide 1.4 Analysis
Core thermal power	3565 MWt
Recirculation sump water volume	9.63×10^4 ft^3
Activity mixed with recirculation loop water	
Noble gases	0.0
Iodines	50% of core inventory
Tritium	97% to sump (water)
Leakage of ECCS equipment outside containment	See Table 6.3-6
Iodine partition factor for leakage	0.1
ABGTS filter efficiencies	
elemental iodine	99%
methyl iodine	99%
particulate iodine	99%
Meteorology	See Table 15.5-14 and Table 15A-2

Table 15.5-13 Doses From Recirculation Loop Leakage Following A LOCA

(rem)	2HR EAB	30 Day LPZ	Control Room
Gamma	4.14E-03	2.28E-02	1.51E-03
Beta	1.36E-03	8.54E-02	1.62E-02
Thyroid - ICRP - 30	1.40E-03	1.53E-01	3.69E-02

Table 15.5-14 Atmospheric Dilution Factors At The Control Building

DILUTION FACTOR (sec/m^3)			
Time Period (hr)	**LOCA/FHA**	**SGTR/MSLB/Loss of AC Power**	**WGDT**
0-2	1.09E-03	2.59E-03	2.56E-03
2-8	9.44E-04	2.12E-03	N/A
8-24	1.56E-04*	N/A	N/A
24-96	1.16E-04**	N/A	N/A
96-720	9.59E-05***	N/A	N/A

GENERAL CONTROL ROOM PARAMETERS

Volume	257,198 cu ft
Makeup/pressurization flow	711 cfm
Recirculation flow	2889 cfm
Unfiltered intake	51 cfm
Filter efficiency	95% first pass
	70% second pass
	0% for noble gases, Tritium
Isolation time, T	40 seconds
Occupancy factors:	
0-24 hr	100%
1-4 days	60%
4-30 days	40%

1. All FHA releases are within 2 hours. Thus, only the 0-2 hr X/Q is applicable for the FHA.

* Calculated value for U1 Shield Bldg Vent to East MCR Intake	1.26E-04
** Calculated value for U1 Shield Bldg Vent to East MCR Intake	9.53E-05
*** Calculated value for U1 Shield Bldg Vent to East MCR Intake	8.07E-05

Table 15.5-15 Deleted by Amendment 97

Table 15.5-16 Parameters Used In Steam Line Break Analysis

	Analysis Value
Steam Generator tube leak rate	
Faulted Steam Generator	1.0 gpm
Per Intact Steam Generator	150 gpd
Iodine Partition Factor	
Faulted Steam Generator	1
Intact Steam Generator	100
RCS Letdown flow rate	124.39 gpm
Steam Releases	
Faulted Steam Generator (0-30 minutes)	96,100 lbm
Three Intact Steam Generators (0-2 hrs)	433,079 lbm
Three Intact Steam Generators (2-8 hrs)	870,754 lbm

Table 15.5-17 Doses From Main Steam Line Break

1 gpm Primary-to-Secondary Leakage (ARCON-96 x/Q)	2 HR EAB	30-Day LPZ	SRP Guidance for 10CFR100 Limits (rem)	Control Room	SRP Guidance for 10CFR100 Limits (rem)
Pre-Accident Initiated Spike Case (14 µCi/gm maximum peak)					
Gamma	2.74E-02	1.11E-02	25	4.32E-03	5
Beta	8.80E-03	4.20E-03	300	3.96E-02	30
Thyroid - ICRP-30	2.41E+00	1.21E+00	300	7.38E+00	30
Accident Initiated Spike Case (0.265 µCi/gm steady state)					
Gamma	1.04E-01	1.25E-01	2.5	8.00E-03	5
Beta	2.54E-02	3.02E-02	30	6.44E-02	300
Thyroid - ICRP-30	3.09E+00	4.78E+00	30	1.03E+01	300

Table 15.5-18 Parameters Used In Steam Generator Tube Rupture Analysis

Primary Side Activity	Technical Specification Limit
Secondary Side Activity	ANSI/ANS-18.1-1984 (Expected levels, 150 gpd/SG)
Iodine Spiking Factor	Case 1: Accident initiated spike of 500 times equilibrium iodine concentration Case 2: Pre-accident spike of 14 µCi/gm I-131 dose equivalent
Iodine Partition Factor	100
Secondary Side Mass Release (Ruptured Steam Generator) 0 -2 hours 2 - 8 hours	 103,300 lbm 32,800 lbm
Secondary Side Mass Release (Intact Steam Generator) 0 - 2 hours 2 - 8 hours	 492,100 lbm 900,200 lbm
Primary Coolant Mass Release (Total) 0 - 2 hours	 191,400 lbm
Primary Coolant Mass Release (Flashed) 0 - 2 hours	 10,077.2 lbm
Meteorology	See Table 15A-2 and 15.5-14

Table 15.5-19 Doses From Steam Generator Tube Rupture

Pre-Accident Initiated Spike Case (14 µCi/gm maximum peak)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	3.78E-01	1.11E-01	6.22E-02
Beta	2.26E-01	6.92E-02	7.01E-01
Thyroid - ICRP-30	1.39E+01	3.79E+00	1.23E+01

Accident Initiated Iodine Spike Case (0.265 µCi/gm steady state)			
(rem)	**2 HR EAB**	**30 DAY LPZ**	**CONTROL ROOM**
Gamma	5.46E-01	1.60E-01	5.71E-02
Beta	2.51E-01	7.73E-02	6.64E-01
Thyroid - ICRP-30	7.19E+00	2.12E+00	2.01E+00

Table 15.5-20 Parameters Used In Fuel Handling Accident Analysis

Regulatory Guide 1.25 Analysis	
Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released	
elemental iodine	99.75%
methyl iodine	0.25%
Filter efficiencies in auxiliary building	RBPVS(2)
elemental iodine	90%
methyl iodine	30%
Amount of mixing of activity in Auxiliary Building	None
Meteorology	See Table 15.5-14 and Table 15A-2

(1) 10% of the total radioactive iodine except for 12% of I-131 and 10% of total noble gases, except for 14% for Kr-85, 5% for Xe-133 and 2% for Xe-135 in the damaged rods at the time of the accident.

(2) Reactor Building Purge Ventilation System

Table 15.5-20a Parameters Used In Fuel Handling Accident Analysis

Regulatory Guide 1.183 Analysis	
Time between plant shutdown and accident	100 hours
Damage to fuel assembly	All rods ruptured
Fuel assembly activity	Highest powered fuel assembly in core region discharged
Activity release to spent fuel pool	Gap activity in ruptured rods(1)
Radial peaking factor	1.65
Form of iodine activity released to spent fuel pool elemental iodine methyl iodine	 99.85%(AST) 0.15%(AST)
Decontamination factor in spent fuel pool	AST Overall=200
Filter efficiencies	No credit taken
Amount of mixing of activity in Auxiliary Building	None
Meteorology	See Table 15.5-14 and Table15A-2

(1) 8% I-131, 10% Kr-85, and 5% other gasses and other halogens.

Table 15.5-21 Nuclear Characteristics Of Highest Rated Discharged Assembly Used In The Analysis

Core thermal power	3565 MWt
Number of assemblies	193
Fuel rods per assembly	264
Core average assembly power	18.47 MWt
Discharged Assembly	
Radial peak to average ratio	1.65

Table 15.5-22 Deleted by Amendment 80

Table 15.5-23
Doses From A Fuel Handling Accident (FHA) (rem)

Doses from Fuel Handling Accident Regulatory Guide 1.183 Analyses

FHA in Auxiliary Building (rem) or In Containment - Containment Open (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	4.29E-01	1.20E-01	5.86E-01
Beta	1.19E+00	3.33E-01	4.68E+00
Thyroid - ICRP-30	5.51E+01	1.54E+01	1.32E+01
TEDE	2.38E+00	6.66E-01	1.02E-00

Doses from Fuel Handling Accident Regulatory Guide 1.25 Analyses

FHA in Reactor Building, Containment Closed (rem)

	2 HR EAB	30 DAY LPZ	CONTROL ROOM
Gamma	4.31E-01	1.21E-01	2.72E-01
Beta	1.24E+00	3.48E-01	2.25E+00
Thyroid - ICRP-30	4.15E+01	1.16E+01	6.81E+00

Table 15.5-24 Deleted by Amendment 80

Table 15.5-25 Deleted by Amendment 80