ERIN

CALCULATION TITLE PAGE

Project No: 122-93-01 Plant Name: Cooper Nuclear Station Calc No: C122-93-01-01

Client Name : NPPD

Subject: CNS Control Room Operator Thyroid Dose Calculation

Computer Program	Standard Computer Program	Program No(s).	Version/Release No.
	☒ YES ☐ NO	PADD	1.1Q

RECORD OF ISSUES

Rev.	Description	Total No. of Sheets	Last Sheet No.	Orig.	Ckd.	App.	Date
0	Initial Issue	See Page 2		/s/ RLD	/s/ MJT	/s/ LAB	4/26/94
1	Credit for mixing eliminated, new source terms.	See Page 2		/s/ RLD	/s/ MJT	/s/ DET for LAB	7/5/94
2	Additional sensitivity cases included.	See Page 2		/s/ RLD	/s/ MJT	/s/ MJT for LAB	7/6/94
3	Additional sensitivity cases included.	See Page 2		/s/ RLD	/s/ MJT	/s/ LAB	7/12/94
4	Run SENSCRIN replaced by three new sensitivity cases.	See Page 2		/s/ RLD	/s/ MJT	/s/ LAB	7/14/94
5	Assumptions revised. New cases included.	See Page 2		RLD	MJT	MJT for LAB	1/9/95

REMARKS

This calculation evaluates the adequacy of the Cooper Nuclear Station Control Room Emergency Ventilation System in mitigating operator thyroid doses following a design basis LOCA or a refueling accident.

Revision 5 includes revisions to pages 1 through 50, and page B-1. Pages 51 through 60 have been added, and Pages B-2 through B-16 have been deleted.

TABLE OF CONTENTS

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE			PAGE
5	RLD	1/9/95	MJT	1/9/95	JOB NO.	122-93-01	2
					CALC NO.	C122-93-01-01	OF
					ERIN® ENGINEERING AND RESEARCH, INC.		60

1.0 PURPOSE

The purpose of this calculation is to update the assumptions, variables and analytical techniques used to calculate the control room operator thyroid, whole body, and beta skin doses following a design basis LOCA and a design basis fuel handling accident. Reference 2 provides the current basis for control room doses due to intake and inleakage of contaminated air. References 1 and 22 provide evaluations of the analysis included in Reference 2. The evaluations identify inconsistencies between plant licensing documentation and the technical bases of the calculations. Recommendations were made in these references to correct inconsistencies and update the analysis through improved analytical techniques. This calculation addresses these issues and provides the updated basis for control room doses, including the incorporation of the Post Accident Design Dose (PADD) computer program.

2.0 SUMMARY OF RESULTS

2.1 SUMMARY OF BASE CASE RESULTS

Control room habitability design basis doses are shown as PADD results in Section 8.0 of this calculation. The design basis or base dose results are presented in this section. Other sensitivity analysis case study results are also presented in Section 8.0. The base case control room operator thyroid, whole body gamma, and beta skin doses were found to be below SRP 6.4 criteria. However, this in itself does not indicate acceptability of the whole body dose with respect to the SRP 6.4 criterion. Whole body doses due to other sources are not evaluated in this calculation. The following table summarizes the results of this analysis given a control room filtered intake flow rate of 1000 CFM and a control room unfiltered inleakage rate of 100 CFM. The refueling case also assumes a 90 second delay in standby gas treatment system actuation.

Event	Dose Type	Calculated Result (rem)	SRP 6.4 Criterion
LOCA	Thyroid	3.91	≤30
	Whole Body Gamma	5.90E-02	≤5
	Beta Skin	5.50E-01	≤30
REFUELING ACCIDENT	Thyroid	1.69	≤30
	Whole Body Gamma	2.67E-01	≤5
	Beta Skin	2.53	≤30

W1229301-5630-010695

2.2 CONCLUSIONS

The Control Room HVAC system at Cooper Nuclear Station (CNS) can meet the Standard Review Plan 6.4 [Reference 34] dose limit of 30 rem thyroid for 30 days following either the design basis loss of coolant accident (LOCA) or the design basis refueling accident described in the CNS Updated Safety Analysis Report (USAR). The whole body doses calculated for CNS are less than the SRP 6.4 criterion of 5 rem. However, as indicated above, additional contributors to the whole body dose must be evaluated to demonstrate compliance with this criterion. The beta skin doses to the operators are also less than the dose limit given above.

This control room dose analysis is deemed to have no impact to existing mitigation actions or protection provided by the control room HVAC system for smoke (internal plant or external plant fire) or hazardous chemical releases as related to control room habitability.

3.0 ASSUMPTIONS

3.1 LOSS OF COOLANT ACCIDENT BASES AND ASSUMPTIONS

General assumptions for the control room operator thyroid and whole body doses calculations for the design basis loss of coolant accident (LOCA) are presented as follows:

1. The reactor has operated for 1000 days at 2381 MWt.

2. Leakage from the primary containment to the reactor building immediately flows through the standby gas treatment system and the stack without mixing in the secondary containment building. However, it is possible to take credit for some mixing (usually ≤50%) within the Reactor Building to reduce the concentration leaving the Reactor Building given prior NRC approval.

3. For normal leakage of radioisotopes from the reactor building, the meteorology assumptions applied in the original CNS disperson coefficient calculation (Reference 42) are applied herein. For the dose contribution due to MSIV leakage, atmospheric dispersion coefficients were calculated based on a release from the turbine building. This derivation is described in Section 8.1.

4. All halogens (iodine and bromine) are assumed to function identically in terms of plateout, filtration, and adsorption.

5. Transport times between the release point and the control room are conservatively assumed as zero.

6. Mixing in the primary containment and control room are assumed to be instantaneous and perfect.

7. The Standby Gas Treatment System (SGTS) filter efficiency is, per CNS Technical Specification 3.7.B.2:

    99% for elemental halogens
    99% for particulate halogens
    99% for organic halogens

As recommended in Reg. Guide 1.52, Rev. 2, and Reference 22, a more conservative value of 95% is used in this calculation for these efficiencies.

8. The control room emergency bypass filter efficiency is, per CNS Technical Specification 3.12.A.2:

   99% for elemental halogens
   99% for particulate halogens
   99% for organic halogens

   As recommended in Reference 22, a more conservative value of 90% is used in this calculation for these efficiencies. This includes a safety factor of 10 to account for degradation between tests. The CNS control room emergency bypass filter does not have humidity control and this factor of safety is consistent with that approved for use at other nuclear power plants with a similar configuration (Reference 41).

9. No credit is taken for plateout of halogens other than that assumed in Regulatory Guide 1.3. It is noted that, if credited, plateout effects would only be applicable to elemental halogens.

10. 100 CFM of unfiltered inleakage into the control room is assumed. This is based on results from test STP 94-199 [Reference 50] (45 CFM ± 26 CFM given 675 CFM filtered flow).

11. For all cases, a one minute delay between the time of initial radioactivity release to the atmosphere and actuation of the control room ventilation emergency bypass mode of operation is assumed. This delay is based on the time which would elapse before the source term is detected at the control room intake and the time requires to actuate the control room emergency ventilation system and align to the filtration mode. There is no operator action or design feature which would preempt this automatic actuation to allow earlier actuation, as is the case with the SGTS. This assumption conservatively increases the source term introduced into the control room envelope during this time period.

12. Upon detection of a low reactor water level, there is an approximate one minute time delay (63 seconds) for realignment from normal containment ventilation to SGTS ventilation. However, as described in NUREG-1465 [Reference 44], actual fuel damage occurs several hours after the start of an event. As this time is well beyond the time required for actuation of the SGTS, it is reasonable to neglect any small delay in SGTS actuation. Therefore, no delay is assumed for the LOCA analyses.

13. Per Reg. Guide 1.3, leakage from primary to secondary containment is assumed to pass directly to the SGTS, where it is then discharged to the environment via the elevated release point (ERP).

14. The control building essential HVAC system has ducting that passes through the CNS control room envelope. This ducting has a source term associated with the noble gas inventory contained therein. However, it displaces an equal quantity of the available source term in the control room and, due to the lack of build-up early in the LOCA event, would likely have a lower concentration of noble gases than the control room. Therefore, it is already accounted for and need not be considered further. In addition, any unfiltered inleakage from this ducting is assumed to be included in the unfiltered flow rates assumed.

15. MSIV leakage is assumed to be held up in the main steam piping for a significant time during transport to the main condenser. A conservative dose reduction factor of 10 is used to account for plateout in the main steam system and condenser (see basis in Appendix A). The radioisotopes are assumed to leak from the condenser at a rate of .5% per day. This leak rate is consistent with the control rod drop dose analysis in Chapter XIV, Section 6.2 of the CNS USAR. The leakage is assumed to immediately pass out of the turbine building to the control room.

16. Per DC 94-102 [Reference 51], a cooling flow of 240 CFM ± 20% is assumed to pass through an inactive SGTS train under accident conditions. For conservatism, a value of 240 CFM + 20%, or 288 CFM, is used. This portion of SGTS flow is filtered at efficiencies of 90% for elemental halogens and 30% for organic halogens. The remaining SGTS flow (1492 CFM) is assumed to pass through the operating SGTS train and be filtered as described in item 7 above.

17. An ECCS leakage contribution of 1000 cc/minute is considered in the post-LOCA dose evaluation. Reference 48 specifies that 10% of the fluid released will flash to vapor and be available for release via the SGTS. This assumption is conservative, because the liquid is subcooled. In addition, historically observed ECCS leakage has been minimal. Walkdowns performed in 1980 for NUREG-0578, Item 2.1.6.a, estimated leakage to be 2 cc/minute. Since 1980, leakage has not noticeably increased and is currently estimated to be minimal. CNS procedures are being revised to monitor ECCS leakage and ensure it is maintained below 1000 cc/minute.

3.2 REFUELING ACCIDENT BASES AND ASSUMPTIONS

The bases and assumptions for the LOCA also apply to the refueling accident with additions and modifications taken from the CNS USAR listed in Section 5.0, Design Inputs, and the following additional assumptions:

1. Noble gas source terms for the refueling accident are derived by multiplying the associated LOCA source terms by an adjustment factor. This factor is the ratio of the number of damaged fuel rods in the refueling accident to the number for a LOCA. From Chapter XIV, Section 6.4.2.1 of the CNS USAR, 125 fuel rods are assumed to fail as a result of a fuel handling accident. From Section 6.3.4, 25% of the fuel rods in the core are assumed to fail as a result of a LOCA. Given 548 fuel bundles of 63 rods each, a total of 8631 rods are assumed to fail following a LOCA. Therefore, the initial LOCA noble gas concentrations are multiplied by a factor of 1.448E-02 to obtain the initial noble gas concentrations for the refueling accident scenario.

2. Halogen source terms are derived in the same way as the noble gas source terms. However, because the LOCA source term assumed only 25% of the halogens are released, an additional factor of 4 was multiplied by the ratio of failed fuel. Also, 99% of the halogens released from the rods is retained by the refueling pool water. Therefore, a partition factor of 100 is used, resulting in a conversion factor for halogens of 5.793E-04.

3. It is assumed there is a 90 second time delay to switch from normal containment ventilation to SGTS ventilation upon detection of a high radiation signal. This is consistent with the reactor building isolation damper closure time of slightly more than one minute (63 seconds). During this time, unfiltered release is assumed to occur, and the reactor building is assumed to remain at a nominal negative pressure.

4.0 METHODOLOGY

The CNS control room operator thyroid and whole body gamma doses are determined based on the analysis of impacts from a design basis LOCA and from a design basis refueling accident. These two accidents are assumed to bound all other accidents with respect to inventory released, source term leakage pathways, and radiological consequences to the control room. The following subsections describe the methodologies used to calculate the resulting thyroid and whole body gamma doses for these accidents.

4.1 GENERAL SUMMARY OF THE LOCA SCENARIO

One of the accidents evaluated to determine the radiation environment for the CNS control room operator thyroid doses is a design basis Loss of Coolant Accident (DBA-LOCA). For CNS the DBA LOCA is defined as a complete circumferential break of one of the reactor recirculation loop lines.

The radiation release path for this scenario is from the reactor vessel to the primary containment. Radiation from the primary containment is assumed to leak to the secondary containment, or reactor building, at maximum allowable rates. Once in the reactor building, the radiation is drawn into the Standby Gas Treatment System (SGTS), treated and released through the stack to the environment. Once in the environment, the material can disperse and potentially be drawn into the control room ventilation system where it can result in operator exposure. The following is a brief description of the events postulated to occur during a DBA-LOCA. The discussion relates to the source term generation and transport of radiation out of containment.

At time t=0, the postulated pipe break occurs and results in rapid blowdown of the reactor coolant system (RCS). Flashing and escape of the coolant during blowdown removes heat rapidly from the primary system and causes the fuel rod cladding temperature to drop. Consequently, only a few fuel rods are assumed to fail during the blowdown period. Following the end of blowdown, the fuel rods are uncovered and the stored heat in the fuel and the decay heat are transferred to the cladding, thus raising the cladding temperature. Some fuel rods may experience cladding failure during this period. The Emergency Core Cooling System (ECCS) refills the lower reactor vessel and then refloods the core region within 100 to 300 seconds, causing cladding temperature to decrease. During the initial blowdown, only the radioactive material contained in the coolant from steady-state operation would be released to the containment. During reflood/refill, when fuel rod cladding failure may occur, the noble gases would be transported out of the reactor system by steam flow and would become airborne. Some fraction of the iodines and less volatile fission products that are released as a result of fuel rod failure would also be transported out of the reactor coolant system by the steam flow and become airborne, and some fraction would remain in solution in the reactor water or would be deposited on surfaces within the reactor coolant system components. The amount that becomes airborne outside the reactor coolant system would be strongly dependent on the time of fuel rod failure and the transport phenomenon for each species within the system.

Following the release from the reactor coolant system, the fission products would be distributed within the drywell or primary containment. The released activity would initially be airborne within the drywell. Following initial release to the containment atmosphere, the high pressure in the primary containment would result in leakage out of the primary containment to the secondary containment. In the secondary containment the action of natural convection currents and ESF equipment, such as cooling fans, will cause time-dependent redistribution of the activity within the secondary containment. Natural removal

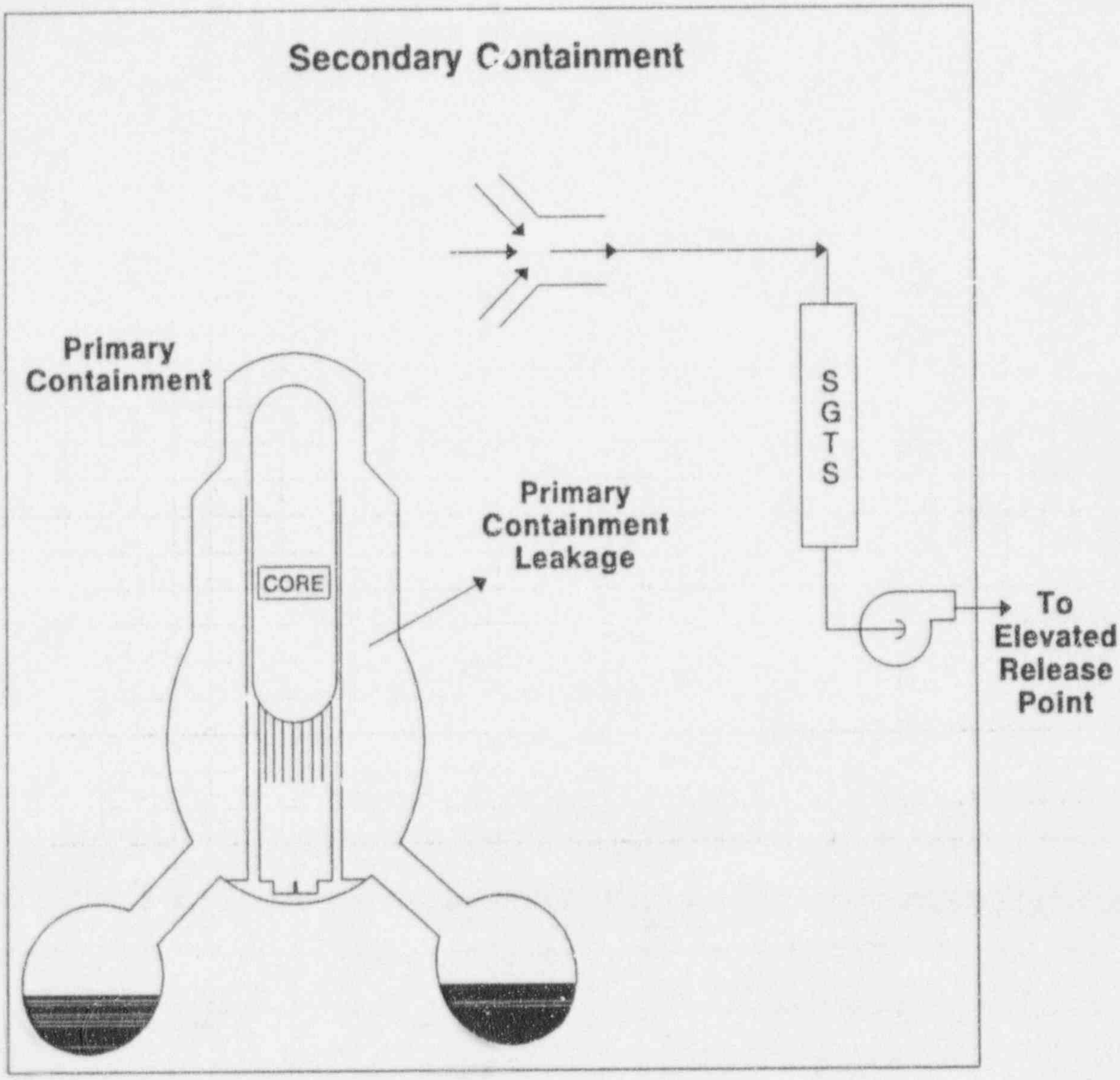
processes, such as deposition on containment surfaces, would reduce the airborne activity concentration and would redistribute a portion of this activity to the containment surfaces.

During the same period of time, leakage of radioactivity from the secondary containment to the atmosphere could take place. This would be processed by the standby gas treatment system (SGTS) filters, causing a buildup of activity on these filters. In addition, there could be some deposition and plateout of radioactivity (iodine and daughters of noble gases) on surfaces of ductwork or on the walls of secondary containment. Figure 4-1 shows the evaluated leakage paths.

During the longer term, contaminated reactor coolant could be circulated through pipes outside of primary containment. Per SRP Section 15.6.5, the Nuclear Regulatory Commission (NRC) staff usually assumes a failure of the seals in the ECCS equipment, such that significant quantities of coolant could leak into compartments outside of containment. At CNS the leaked fluid is either retained in the room or transported to the radwaste system. Some portion of this leaked fluid is volatilized and also transported in the air of these compartments. These sources would be processed by the SGTS filters prior to transport to the control room.

Figure 4-1

COOPER NUCLEAR STATION POST-LOCA SOURCE TERM LEAKAGE PATHWAYS



W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	RLD	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE
					CALC NO. C122-93-01-01	13
					ERIN® ENGINEERING AND RESEARCH, INC.	OF 60

4.2 MAIN STEAM ISOLATION VALVES

The Cooper Nuclear Station (CNS) main steam system design includes Main Steam Isolation Valves (MSIVs) that provide containment isolation capability for the main steam lines where they exit the primary containment. These valves are arranged in series in four, twenty-four inch steam lines for a total of eight (8) valves. Each steam line has two MSIVs, one inside and one outside the containment barrier. The MSIVs are spring loaded, pneumatic piston-operated globe valves designed to fail closed on loss of pneumatic pressure or loss of power to the pilot valves. Each valve has an air accumulator to assist in the closure of the valve upon loss of the air supply, electrical power to the pilot valves, and failure of the loaded spring. During a DBA LOCA the MSIVs would close upon low water level in the reactor vessel. Once isolation is initiated, the valves will continue to close and cannot be opened except by deliberate manual operator action.

During a DBA LOCA the MSIVs function to isolate main steam following a plant trip and, among other things, form a portion of the primary containment boundary for post-LOCA retention of the radionuclide inventory. As part of the CNS post-LOCA control room operator thyroid dose calculation, it has been determined that some contribution of the source term will bypass the primary/secondary containment features via design basis leakage through these valves. A technical basis has been developed for the consideration of retention of much of the released source term in the BOP systems. This technical evaluation is provided as Appendix A to this calculation. Based on this evaluation, a conservative dose reduction factor of 10 is used in this evaluation.

As described in Section 3.1, radioactive material leaking through the main steam lines is transported to the main condenser, where it is then released to the turbine building and then to the environment. This alternate release point requires the application of different

atmospheric dispersion coefficients. However, this cannot be done concurrently with the evaluation of normal containment leakage. Therefore, for each LOCA scenario considered, two PADD runs were performed. One run determines the doses due to primary containment leakage exclusive of MSIV leakage, and the other evaluates the consequences of MSIV leakage only. The resulting doses were then added to obtain the total doses for each scenario.

4.3 OTHER LOCA SCENARIO CONSIDERATIONS

4.3.1 Revised Halogen Filter Efficiencies and Partitioning Factors

Following a loss of coolant accident, normal reactor building ventilation will be isolated, and one train of the standby gas treatment system will start. The design flow rate of this system is 1780 CFM. However, as described in DC 94-102, 240 CFM ± 20% is assumed to pass through the inactive train. For conservatism, this is assumed to be 240 CFM + 20%, or 288 CFM. In accordance with Regulatory Guide 1.52 [Reference 16], this flow is assumed to be filtered at an efficiency of 90% for elemental halogens and 30% for organic halogens. The remaining flow of 1492 CFM is assumed to pass through the operating SGTS train with filter efficiencies as described in Section 3.1. For both cases, particulate halogens are assumed to be filtered at an efficiency of 95%. However, PADD cannot explicitly model this scenario. Therefore, the SGTS filter efficiency and halogen partitioning factors (see item 6, Section 5) must be adjusted to account for this. Figure 4-2 illustrates the process for calculating the revised filter efficiency and partitioning factors. For example, consider the organic halogens passing through the SGTS. According to Regulatory Guide 1.3, 4% of the released halogens are in the organic form. This results in effective rates of organic halogen discharge of 59.68 CFM through the operating SGTS train and 11.52 CFM through the inactive train. Given filter efficiencies of 95% and 30%, respectively, essentially 2.984 CFM of organic halogens is being

released through the operating train, while 8.064 CFM is being released through the inactive train. Similarly, for elemental halogens, given a partitioning factor of 91% and filter efficiencies of 95% and 90%, 67.886 CFM and 26.208 CFM will be released from the active and inactive trains, respectively. For particulate halogens, the filter efficiency is 95% in both cases, resulting in a total of 4.45 CFM being released. The total volumetric flow rate of released halogens is 109.952 CFM. Comparing this value to the original total flow rate of 1780 CFM yields an effective filter efficiency of 93.84%. The updated partitioning factors are calculated as follows:

- Elemental halogens

  (67.886 + 26.208) / 109.952 = .8586

- Particulate halogens

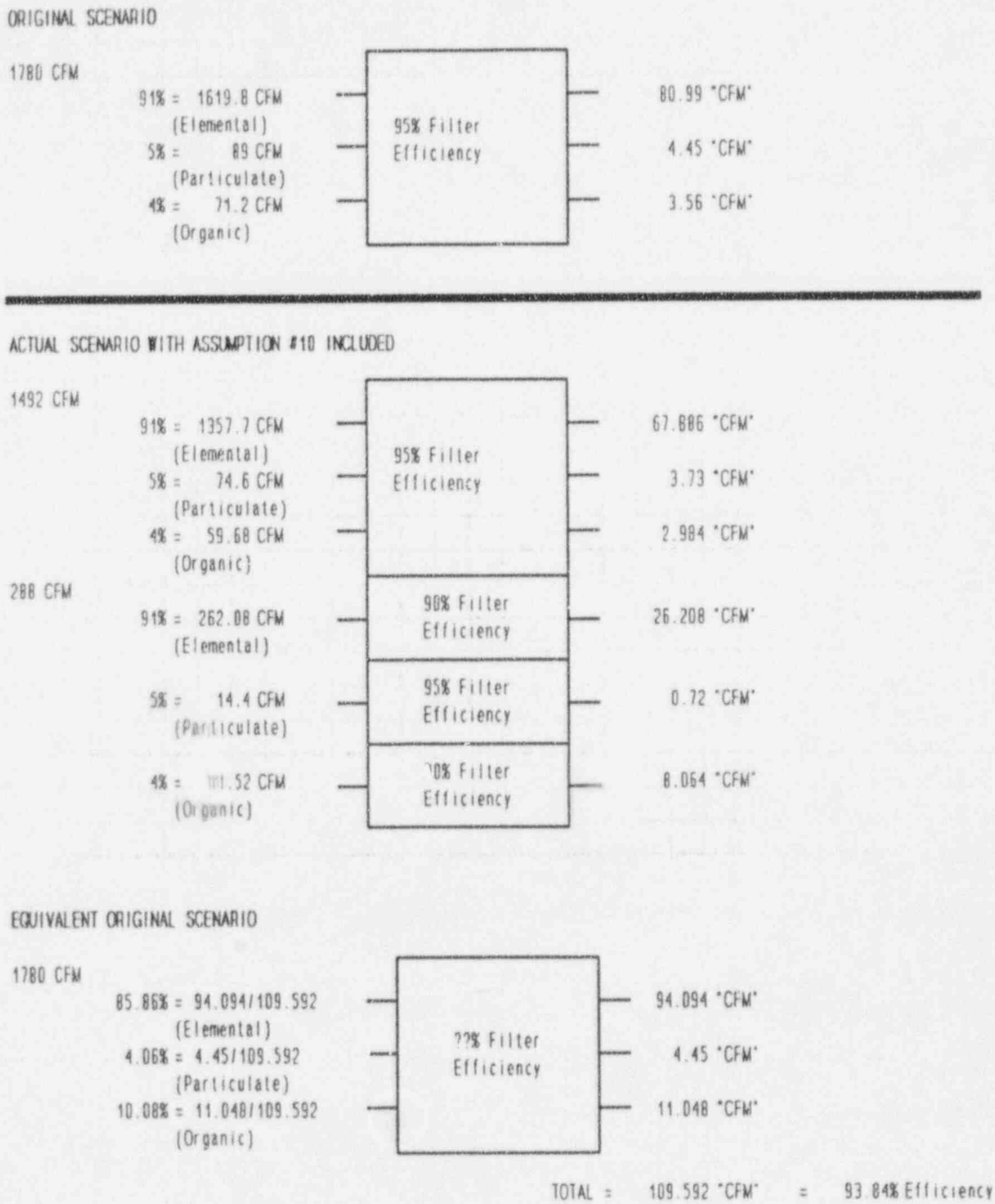
  4.45 / 109.952 = .0406

- Organic halogens

  (2.984 + 8.064) / 109.952 = .1008

Figure 4-2

CALCULATION OF EFFECTIVE FILTER EFFICIENCY AND PARTITIONING FACTORS



W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	[signature]	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE
					CALC NO. C122-93-01-01	17
					ERIN®	OF
					ENGINEERING AND RESEARCH, INC.	60

4.3.2 ECCS Leakage

As a result of evaluations of leakage from the ECCS recirculation system, it was necessary to include the dose contribution due to ECCS leakage. This is accomplished by revising the primary containment leak rate to include this additional contribution. However, because the ECCS leakage involves fluid leakage rather than vapor leakage, the leak rate was corrected for this. From Reference 31, 25% of the halogen inventory available for release is released to the primary containment atmosphere, and from Reference 48, an additional 50% is mixed with the ECCS water volume. Given a water volume of 96,445 ft^3 [Reference 49] and assuming a release factor of 0.1 (see Section 3.1, Item 17), an ECCS leak rate of 1000 cc/min is equivalent to a halogen leak rate of 0.00264% of the total halogens available for release per day. Given an overall primary containment leak rate of 0.635% per day, the corresponding leak rate from the primary containment atmosphere is 0.1588% of the available halogens per day. Taking the ratio of the sum of the primary containment (0.1588%) and ECCS (0.00264%) halogen vapor leak rates to the primary containment halogen vapor leak rate alone and multiplying by the original primary containment leak rate of 0.635% per day results in an adjusted primary containment leak rate of 0.646% per day.

4.4 GENERAL SUMMARY OF THE REFUELING ACCIDENT SCENARIO

Accidents that result in the release of radioactive materials directly to the containment can occur when the drywell is open. A survey of the various conditions that could exist when the drywell is open reveals that the greatest potential for the release of radioactive material occurs when the drywell head and reactor vessel head have been removed. In this case, radioactive material released as a result of fuel failure is available for transport directly to the containment.

Various mechanisms for fuel failure under this condition have been investigated. With the current fuel design, the refueling interlocks, which impose restrictions on the movement of refueling equipment and control rods, prevent an inadvertent criticality during refueling operations. In addition, the reactor protection system can initiate a reactor scram in time to prevent fuel damage for errors or malfunctions occurring during planned criticality tests with the reactor vessel head off. It is concluded that the only accident that could result in the release of significant quantities of fission products to the containment during this mode of operation is one resulting from the accidental dropping of a fuel bundle onto the top of the core.

This event occurs under non-operating conditions for the fuel. The key assumption of this postulated occurrence is the inadvertent mechanical damage to the fuel rod cladding as a consequence of the fuel bundle being dropped on the core while in the cold condition. Fuel densification considerations do not enter into or affect the accident results.

4.5 CONTROL ROOM ACTIVITY CALCULATION EQUATIONS

The calculations presented in this study are performed with the use of the PADD computer program. PADD models a three compartment system of volumes. Typically, compartment 1 represents the primary containment, compartment 2 represents the reactor building or secondary containment, and compartment 3 represents the control room. PADD calculates airborne isotopic radioactivity concentrations as a function of time in each of the three compartments, taking into account removal due to radioactive decay and leakage out of the compartment, as well as production due to leakage into the compartment. The postulated DBA releases consist almost exclusively of halogens and noble gases. Due to the neutron-rich nature of fission products, decay tends toward isotopes with higher atomic numbers. In other words, halogens tend to decay into noble gases, which then decay into other isotopes. Therefore, production of halogens from radioactive decay of other isotopes does not occur, although production of noble gases from radioactive decay of halogen isotopes does.

Using the information described above, PADD then calculates dose rates and integrated doses for the control room and two offsite locations specified by the user. Meteorological data modeling atmospheric dispersion is also an input to the code.

The airborne isotopic radioactivity concentrations are based upon closed form solutions of the differential equations describing production and removal of isotopes in each compartment. Dose rates are calculated by applying appropriate dose conversion factors to the airborne radioactivity concentrations. Integrated doses are calculated numerically using the rectangle rule. The rectangle rule estimates the value of an integral during a time step by assuming the integral is constant over the time interval. This constant is equal to the value of the integral at the midpoint of the interval. The specific equations employed within the PADD program are presented as follows:

4.5.1 Halogen Concentration in Primary Containment

$$\underset{[t_{N-1} \le t \le t_N]}{C_{1H}(t)} = \frac{S_{iH}(t)}{V_1} \left\{ \left[e^{-\sum_{\kappa=1}^{N-1} \left[\frac{(Q_1 + Q_\ell)}{V_1} \right]_\kappa (t_\kappa - t_{\kappa-1})} e^{-\left[\frac{(Q_1 + Q_\ell)}{V_1} \right]_N (t - t_{N-1})} \right] \left[f_e e^{-\sum_{\kappa=1}^{N-1} \lambda_{p\kappa} (t_\kappa - t_{\kappa-1})} e^{-\lambda_{pN}(t - t_{N-1})} + f_p + f_o \right] \right\} \tag{1}$$

4.5.2 Noble Gas Concentration in Primary Containment

$$\underset{[t_{N-1} \le t \le t_N]}{C_{1N}(t)} = e^{-\tau_N (t - t_{N-1}) - \sum_{\kappa=1}^{N-1} \tau_\kappa (t_\kappa - t_{\kappa-1})} \left\{ \left[\frac{S_{iN}(t)}{V_1} + e^{-\lambda_i (t - t_{N-1})} \sum_j \left[\frac{\lambda_j S_{jH}(t)}{V_1(\lambda_i - \lambda_j)} \left(e^{-\lambda_i (t - t_{N-1})} - e^{\lambda_j (t - t_{N-1})} \right) \right] \right] \right.$$

$$\left. + \sum_{\kappa=1}^{N-2} \left[e^{-\lambda_i (t - t_{N-1}) - \sum_{p=\kappa+1}^{N-1} \lambda_i (t_p - t_{p-1})} \sum_j \left(\frac{\lambda_j S_{jH}(t_{N-1})}{V_1(\lambda_i - \lambda_j)} \left[e^{-\lambda_i (t_{\kappa+1} - t_\kappa)} - e^{\lambda_j (t_{\kappa+1} - t_\kappa)} \right] \right) \right] \right\} \tag{2}$$

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	RJD	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE
					CALC NO. C122-93-01-01	21
					ERIN® ENGINEERING AND RESEARCH, INC.	OF 60

4.5.3 Halogen Concentration in Secondary Containment

$$
\begin{aligned}
\underset{[t_{N-1} \le t \le t_N]}{C_{2_H}(t)} = {} & \frac{S_{iH}(t)}{V_1}\left[\frac{Q_1}{V_2}\right]_N e^{-\sum_{r=1}^{N-1} r_r (t_r - t_{r-1})} \\
& \times \left[\frac{f_p + f_o}{\gamma_N}\left(e^{-r_N (t - t_{N-1})} - e^{-\epsilon_N (t - t_{N-1})}\right) + \frac{f_e e^{-\sum_{r=1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_{N-1} - \lambda_{pN}}\left(e^{-(r_N + \lambda_{pN})(t - t_{N-1})} - e^{-\epsilon_N (t - t_{N-1})}\right)\right]
\end{aligned}
\quad (3)
$$

$$
\begin{aligned}
& + e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})} \times \left\{ \frac{S_{iH}(t_{N-1})}{V_1}\left[\frac{Q_1}{V_2}\right]_{N-1} e^{-\sum_{r=1}^{N-2} r_r (t_r - t_{r-1})} \right. \\
& \left. \times \left[\frac{f_p + f_o}{\gamma_{N-1}}\left(e^{r_{N-1}(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})}\right) + \frac{f_e e^{-\sum_{r=1}^{N-2} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_{N-1} - \lambda_{pN-1}}\left(e^{-(r_{N-1} + \lambda_{pN-1})(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})}\right)\right]\right\}
\end{aligned}
$$

$$
+ \sum_{K=1}^{N-2}\left\{\left(\frac{Q_1}{V_2}\right)_K \frac{S_{iH}(t_K)}{V_1} e^{-\sum_{r=K+1}^{N-1}(\epsilon_r + \lambda_i)(t_r - t_{r-1}) - \sum_{r=1}^{K-1} r_r (t_r - t_{r-1})} \left[\frac{f_p + f_o}{\gamma_K}\left(e^{-r_K (t_K - t_{K-1})} - e^{-\epsilon_K (t_K - t_{K-1})}\right) + f_e \frac{e^{-\sum_{r=1}^{K-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_K - \lambda_{pK}}\left(e^{-(r_K + \lambda_{pK})(t_K - t_{K-1})} - e^{-\epsilon_K (t_K - t_{K-1})}\right)\right]\right\}\Bigg\}
$$

REV.	BY	DATE	CHECKED	DATE		
5	[signature]	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE 22
					CALC NO. C122-93-01-01	OF 60

4.5.4 Noble Gas Concentration in Secondary Containment

$$
\begin{aligned}
\underset{t_{N-1} \le t \le t_N}{C_{2_N}(t)} = & \left(\frac{Q_1}{V_2}\right)_N T_1 + e^{-(\epsilon_N+\lambda_i)(t-t_{N-1})} \\
& \times \left\{ \left(\frac{Q_1}{V_2}\right)_{N-1} T_2 + \sum_{K=1}^{N-2} \left[\left(\frac{Q_1}{V_2}\right)_K e^{-\sum_{r=K+1}^{N-1} (\epsilon_r+\lambda_i)(t_r-t_{r-1})} T_{3_K} \right] \right\} \\
& + \sum_j \left\{ \lambda_j \left[T_4 + e^{-(\epsilon_N+\lambda_i)(t-t_{N-1})} \left[T_5 + \sum_{K=1}^{N-2} \left(e^{-\sum_{r=K+1}^{N-1} (\epsilon_r+\lambda_i)(t_r-t_{r-1})} T_{6_K} \right) \right] \right] \right\}
\end{aligned}
\tag{4}
$$

Where:

$$T_1 \equiv e^{-(\epsilon_N+\lambda_i)\,t} \left[I_{1_N} - I_{1_N}\big|_{t_{N-1}} \right] \tag{4a}$$

$$T_2 \equiv e^{-(\epsilon_{N-1}+\lambda_i)\,t_{N-1}} \left[I_{1_{N-1}}\big|_{t_{N-1}} - I_{1_{N-1}}\big|_{t_{N-2}} \right] \tag{4b}$$

$$T_{3_K} \equiv e^{-(\epsilon_K+\lambda_i)\,t_K} \left[I_{1_K}\big|_{t_K} - I_{1_K}\big|_{t_{K-1}} \right] \tag{4c}$$

$$T_4 \equiv e^{-(\epsilon_N+\lambda_i)\,t} \left[I_{2_N} - I_{2_N}\big|_{t_{N-1}} \right] \tag{4d}$$

$$T_5 \equiv e^{-(\epsilon_{N-1}+\lambda_i)\,t_{N-1}} \left[I_{2_{N-1}}\big|_{t_{N-1}} - I_{2_{N-1}}\big|_{t_{N-2}} \right] \tag{4e}$$

$$T_{6_K} \equiv e^{-(\epsilon_K+\lambda_i)\,t_K} \left[I_{2_K}\big|_{t_K} - I_{2_K}\big|_{t_{K-1}} \right] \tag{4f}$$

4.5.5 Halogen Concentration in Control Room

$$\begin{aligned} \underset{[t_{N-1} \le t \le t_N]}{C_{3_N}(t)} &= T_{2_N}(t) - e^{-(K_{i_N} + \lambda_i)(t - t_{N-1})}\, T_{2_N}(t_{N-1}) \\ &\quad + T_{3_N}(t) - e^{-(K_{i_N} + \lambda_i)(t - t_{N-1})}\, T_{3_N}(t_{N-1}) \\ &+ \sum_{K-1}^{N-1} \left\{ e^{-\sum_{r=K+1}^{N} (K_{i_r}+\lambda_i)(t_r - t_{r-1})} \left[\left(T_{2_K}(t_K) - e^{-(K_{i_K}+\lambda_i)(t_K - t_{K-1})}\, T_{2_K}(t_{K-1}) \right) + \left(T_{3_K}(t_K) - e^{-(K_{i_K}+\lambda_i)(t_K - t_{K-1})}\, T_{3_K}(t_{K-1}) \right) \right] \right\} \end{aligned} \tag{5}$$

Where:

$$\begin{aligned} T_2(t) &= E_{3_{I_N}} \frac{X}{Q_N} (1 - e_{2_{I_N}})\, Q_{2_N} \Big\{ \frac{S_{IH}(t)}{V_1} \left(\frac{Q_1}{V_2}\right)_N e^{-\sum_{r-1}^{N-1} \tau_r (t_r - t_{r-1})} \Big[\frac{f_p + f_o}{\gamma_N} \left(\frac{e^{-\tau_N (t - t_{N-1})}}{\delta_{IN}} - \frac{e^{-\epsilon_N (t - t_{N-1})}}{\theta_{IN}} \right) \\ &\quad + \frac{f_e e^{-\sum_{r-1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_N - \lambda_{pN}} \left(\frac{e^{-(\tau_N + \lambda_{pN})(t - t_{N-1})}}{\delta_{IN} - \lambda_{pN}} - \frac{e^{-\epsilon_N (t - t_{N-1})}}{\theta_{IN}} \right) \Big] + \frac{e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Big[\frac{S_{IH}(t_{N-1})}{V_1} \\ &\quad \times \left(\frac{Q_1}{V_2}\right)_{N-1} e^{-\sum_{r=1}^{N-2} \tau_r (t_r - t_{r-1})} \Big[\frac{f_p + f_o}{\gamma_{N-1}} \left(e^{-\tau_{N-1}(t_{N-1} - t_{N-2})} - e^{\epsilon_{N-1}(t_{N-1} - t_{N-2})} \right) + \frac{f_e\, e^{-\sum_{r=1}^{N-2} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_{N-1} - \lambda_{pN-1}} \\ &\quad \times \left(e^{-(\tau_{N-1} + \lambda_{pN-1})(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})} \right) \Big] + \sum_{K=1}^{N-2} \Big\{ \left(\frac{Q_1}{V_2}\right)_K \frac{S_{IH}(t_K)}{V_1} e^{-\sum_{r=K+1}^{N-1} (\epsilon_r + \lambda_i)(t_r - t_{r-1})} \\ &\times e^{-\sum_{r=1}^{K-1} \tau_r (t_r - t_{r-1})} \left[\frac{f_p + f_o}{\gamma_K} \left(e^{-\tau_K (t_K - t_{K-1})} - e^{-\epsilon_K (t_K - t_{K-1})} \right) + \frac{f_e\, e^{-\sum_{r=1}^{K-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_K - \lambda_{pK}} \left(e^{-(\tau_K + \lambda_{pK})(t_K - t_{K-1})} - e^{\epsilon_K (t_K - t_{K-1})} \right) \right. \end{aligned} \tag{6}$$

(Note that $\delta_{iK} \equiv K_{iK} - T_K$ and $\theta_{iK} \equiv K_{iK} - \epsilon K_K$)

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	PLD	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE 24
					CALC NO. C122-93-01-01	OF 60
					ERIN® ENGINEERING AND RESEARCH, INC.	

$$T_3 = E_{3_i} \frac{X}{Q_G} Q_E \times \frac{S_{IH}(t)}{V_1} e^{-\tau_N(t-t_{N-1}) - \sum_{r=1}^{N-1} \tau_r(t_r - t_{r-1})} \times \left[\frac{f_e \, e^{-\lambda_{pN}(t-t_{N-1}) - \sum_{r=1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{\delta_{IN} - \lambda_{pN}} + \frac{f_p + f_o}{\delta_{IN}} \right] \quad (7)$$

(Note that $\delta \equiv K - T$).

4.5.6 Noble Gas Concentration in Control Room

$$\begin{aligned} \underset{[t_{N-1} \le t \le t_N]}{C_{3_N}(t)} &= \left[T_{2_N}(t) - e^{-(K_{IN}+\lambda_i)(t-t_{N-1})} T_{2_N}(t_{N-1}) \right] + \left[T_{3_N}(t) - e^{-(K_{IN}+\lambda_i)(t-t_{N-1})} T_{3_N}(t_{N-1}) \right] \\ &+ \left[T_{4_N}(t) - e^{-(K_{IN}+\lambda_i)(t-t_{N-1})} T_{4_N}(t_{N-1}) \right] + \sum_{K=1}^{N-1} \left\{ e^{-\sum_{r=K+1}^{N} (K_{Ir}+\lambda_i)(t_r - t_{r-1})} \right. \\ &\times \left[\left(T_{2_K}(t) - e^{-(K_{IK}+\lambda_i)(t_K - t_{K-1})} T_{2_K}(t_{K-1}) \right) + \left(T_{3_K}(t) - e^{-(K_{IK}+\lambda_i)(t_K - t_{K-1})} T_{3_K}(t_{K-1}) \right) \right. \\ &\left. \left. + \left(T_{4_K}(t) - e^{-(K_{IK}+\lambda_i)(t_K - t_{K-1})} T_{4_K}(t_{K-1}) \right) \right] \right\} \end{aligned} \quad (8)$$

$$T_{2_N}(t) = E_{3_N}\left[\frac{\chi}{Q_G}\right]_N Q_{E_N}\, e^{-\tau_N(t-t_{N-1})-\sum_{n=1}^{N-1}\tau_n(t_n-t_{n-1})} \Bigg\{ \frac{S_{iN}(t)}{V_i\delta_{iN}}$$

$$+\, e^{-\lambda_i(t-t_{N-1})} \sum_j \left[\frac{\lambda_j S_{jH}(t)}{V_i\,(\lambda_i-\lambda_j)}\left[\frac{e^{-\lambda_i(t-t_{N-1})}}{\delta_{in}-\lambda_i-\lambda_j} - \frac{e^{\lambda_i(t-t_{N-1})}}{\delta_{iN}}\right]\right] \qquad (9)$$

$$+\, \sum_{\kappa=1}^{N-2}\left[e^{\lambda_i(t-t_{N-1})-\sum_{r=\kappa+1}^{N-1}\lambda_i(t_r-t_{r-1})}\sum_j\left(\frac{\lambda_j S_{jH}(t_{N-1})}{\delta_{iN}V_1(\lambda_i-\lambda_j)}\right.\right.$$

$$\left.\left.\times\,\left\{e^{-\lambda_i(t_{\kappa+1}-t_\kappa)} - e^{\lambda_i(t_{\kappa+1}-t_\kappa)}\right\}\right)\right]\Bigg\}$$

$$T_3(t) = E_{3_{iN}}\left[\frac{\chi}{Q}\right]_N \left(1-e_{2_{iN}}\right)Q_{2_N}\, e^{-(K_{iN}+\lambda_i)\,t}\,\{T_1^*(t)+T_2^*(t)+T_3^*(t)+T_4^*(t)+T_5^*(t)+T_6^*(t)\} \qquad (10)$$

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	RJD	1/9/95	MDT	1/9/95	JOB NO. 122-93-01	PAGE
					CALC NO. C122-93-01-01	26
					ERIN®	OF
					ENGINEERING AND RESEARCH, INC.	60

$$
\begin{aligned}
T_1^*(t) = {} & e^{-\sum_{r=1}^{N-1} \epsilon_r (t_r - t_{r-1})} \, e^{(K_{IN}+\lambda_I)t} \left(\frac{Q_1}{V_2}\right)_N \left\{ \frac{1}{V_1 \gamma_N} \left[\frac{S_{IN}(t)\, e^{-\epsilon_N (t - t_{N-1})}}{\delta_{IN}} - \frac{S_{IN}(t_{N-1})\, e^{-(\epsilon_N + \lambda_I)(t - t_{N-1})}}{\theta_{IN}} \right] \right. \\
& + \sum_J \left[\frac{\lambda_I}{V_1(\lambda_I - \lambda_J)} \left(\frac{1}{\gamma_N - \lambda_J - \lambda_I} \left\{ \frac{S_{JH}(t)\, e^{-(\epsilon_N + 2\lambda_I)(t - t_{N-1})}}{\delta_{IN} - \lambda_J - \lambda_I} - \frac{S_{JH}(t_{N-1})\, e^{(\epsilon_N + \lambda_I)(t - t_{N-1})}}{\theta_{IN}} \right\} \right. \right. \\
& \left. \left. - \frac{1}{\gamma_N} \left\{ \frac{S_{JH}(t)\, e^{-(\epsilon_N + \lambda_I + \lambda_J)(t - t_{N-1})}}{\delta_{IN} - 2\lambda_I} - \frac{S_{JH}(t_{N-1})\, e^{-(\epsilon_N + \lambda_I)(t - t_{N-1})}}{\theta_{IN}} \right\} \right) \right] \\
& + \left[\frac{e^{-\lambda_I (t - t_{N-1})}}{K_{IN}} - \frac{e^{-(\epsilon_N + \lambda_I)(t - t_{N-1})}}{\theta_{IN}} \right] \sum_{r=1}^{N-2} \left[e^{-\sum_{Z=r+1}^{N-1} \lambda_I (t_Z - t_{Z-1})} \sum_J \left(\frac{\lambda_J S_{JH}(t_{N-1})}{V_1(\lambda_I - \lambda_J)} \right. \right. \\
& \left. \left. \left. \times \left\{ \frac{e^{-\lambda_I (t_{r+1} - t_r)} - e^{-\lambda_J (t_{r+1} - t_r)}}{\gamma_N - \lambda_I} \right\} \right) \right] \right\}
\end{aligned}
\tag{10a}
$$

$$
T_2^*(t) = \left(\frac{Q_1}{V_2}\right)_{N-1} T_2' \, \frac{e^{(K_{IN}+\lambda_I)t - (\epsilon_N + \lambda_I)(t - t_{N-1})}}{\theta_{IN}}
\tag{10b}
$$

$$
T_3^*(t) = \sum_{K=1}^{N-2} \left\{ \left(\frac{Q_1}{V_2}\right)_K \frac{e^{(K_{IN}+\lambda_I)t - \sum_{r=K+1}^{N-1} (\epsilon_r + \lambda_I)(t_r - t_{r-1})}}{K_{IN} + \lambda_I} \, T_{3_K}' \right\}
\tag{10c}
$$

W1229301-5630-010695

5	PJD	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE
					CALC NO. C122-93-01-01	27
					ERIN®	OF
REV.	BY	DATE	CHECKED	DATE	ENGINEERING AND RESEARCH, INC.	60

$$
\begin{aligned}
T_4'(t) = {} & e^{(K_{IN}+\lambda_i)t}\left(\frac{Q_1}{V_2}\right)_N \frac{e^{-\sum_{r=1}^{N-1} \tau_r (t_r - t_{r-1})}}{V_1} \Bigg\{ \frac{f_p + f_o}{\gamma_N} \Bigg[\frac{1}{\gamma_N + \lambda_i - \lambda_j} \Bigg(\frac{S_{jN}(t)\, e^{-\tau_N (t - t_{N-1})}}{\delta_{IN} + \lambda_i - \lambda_j} \\
& - \frac{S_{jH}(t_{N-1})\, e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg) - \frac{1}{\lambda_i - \lambda_j} \Bigg(\frac{S_{jH}(t)\, e^{-\epsilon_N (t - t_{N-1})}}{\theta_{IN} + \lambda_i - \lambda_j} - \frac{S_{jN}(t_{N-1})\, e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg) \Bigg] \\
& + \frac{f_e e^{-\sum_{r=1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_N - \lambda_{p_N}} \Bigg[\frac{1}{\gamma_N - \lambda_{p_N} + \lambda_i - \lambda_j} \Bigg(\frac{S_{jH}(t)\, e^{-(\tau_N + \lambda_{pN})(t - t_{N-1})}}{\delta_{IN} - \lambda_{p_N} + \lambda_i - \lambda_j} - \frac{S_{jH}(t_{N-1})\, e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg) \\
& - \frac{1}{\lambda_i - \lambda_j} \Bigg(\frac{S_{jN}(t)\, e^{-\epsilon_N (t - t_{N-1})}}{\delta_{IN} + \lambda_i - \lambda_j} - \frac{S_{jH}(t_{N-1})\, e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg) \Bigg] \Bigg\} + \Bigg\{ \frac{e^{-(\epsilon_N + \lambda_j)(t - t_{N-1})}}{\theta_{IN} + \lambda_i - \lambda_j} - \frac{e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg\} \left(\frac{1}{\lambda_i - \lambda_j} \right) \\
& \times \Bigg\{ \left(\frac{Q_1}{V_2} \right)_{N-1} S_{jH}(t_{N-1})\, e^{-\sum_{r=1}^{N-2} \tau_r (t_r - t_{r-1})} \Bigg\{ \frac{f_p + f_o}{\gamma_{N-1}} \left[e^{-\tau_{N-1}(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})} \right] + \frac{f_e e^{-\sum_{r=1}^{N-2} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_{N-1} - \lambda_{p_{N-1}}} \\
& \times \left[e^{-(\tau_{N-1} + \lambda_{p_{N-1}})(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})} \right] \Bigg\} + \Bigg\{ \frac{1}{K_{IN} + \lambda_i} - \frac{e^{-(\epsilon_N + \lambda_i)(t - t_{N-1})}}{\theta_{IN}} \Bigg\} \sum_{\ell=1}^{N-2} \Bigg\{ \left(\frac{Q_1}{V_2} \right)_\ell \frac{S_{jH}(t_\ell)}{V_1} \\
& \times e^{-\sum_{r=\ell+1}^{N-1} (\epsilon_r + \lambda_i)(t_r - t_{r-1}) - \sum_{r=1}^{\ell-1} \tau_r (t_r - t_{r-1})} \Bigg[\frac{f_p + f_o}{\gamma_\ell} \left(e^{-\tau_\ell (t_\ell - t_{\ell-1})} - e^{-\epsilon_\ell (t_\ell - t_{\ell-1})} \right) + \frac{f_e e^{-\sum_{r=1}^{\ell-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_\ell - \lambda_{p\ell}} \\
& \times \left(e^{-(\tau_\ell + \lambda_{p\ell})(t_\ell - t_{\ell-1})} - e^{-\epsilon_\ell (t_\ell - t_{\ell-1})} \right) \Bigg] \Bigg\} \Bigg\}
\end{aligned}
\tag{10d}
$$

$$
T_5'(t) = T_5' \frac{e^{(K_{IN} + \lambda_i)t - (\epsilon_{N-1} + \lambda_i)(t - t_{N-1})}}{\theta_{IN}}
\tag{10e}
$$

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE				
5	RHD	1/9/95	MJT	1/9/95	JOB NO.	122-93-01	PAGE	28
					CALC NO.	C122-93-01-01	OF	60

ERIN®
ENGINEERING AND RESEARCH, INC.

$$T_6'(t) = \sum_{K=1}^{N-2} \left\{ \frac{e^{(K_{iN}+\lambda_i)t - \sum_{r=K+1}^{N-1} (\epsilon_r+\lambda_i)(t_r - t_{r-1})}}{K_{iN} + \lambda_i} T_{6_K}' \right\}$$

(10f)

Where

$$T_4(t) = \sum_j \left\{ \lambda_j \left[\left[I_{41}' - \frac{e^{-(K_{iN}+\lambda_i)(t-t_{N-1})}}{K_{iN}-K_{jN}+\lambda_i-\lambda_j} T_{2_N}'(t_{N-1}) + I_{42}' - \frac{e^{-(K_{iN}+\lambda_i)(t-t_{N-1})}}{K_{iN}-K_{jN}+\lambda_i-\lambda_j} T_{3_N}'(t_{N-1}) + I_{43}' \right] \right] \right\}$$

(11)

W1229301-5630-010695

$$I'_{41} = E_{3jN} \left(\frac{X}{Q}\right)_N (1-e_{2jN})\ Q_{2N} \Bigg\{ \frac{S_{jH}(t)}{V_1} \left(\frac{Q_1}{V_2}\right)_N e^{-\sum_{r=1}^{N-1} \tau_r(t_r - t_{r-1})}$$

$$\times \Bigg[\frac{f_p + f_o}{\gamma_N} \left(\frac{e^{-\tau_N(t-t_{N-1})}}{\delta_{jN}(\delta_{IN} + \lambda_I - \lambda_j)} - \frac{e^{-\epsilon_N(t-t_{N-1})}}{\theta_{jN}(\theta_{IN} + \lambda_I - \lambda_j)} \right) + \frac{f_e\, e^{-\sum_{r=1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_N - \lambda_{pN}} \Bigg(\frac{e^{-(\tau_N + \lambda_{pN})(t-t_{N-1})}}{(\delta_{jN} - \lambda_{pN})(\delta_{IN} - \lambda_{pN}}$$

$$- \frac{e^{-\epsilon_N(t-t_{N-1})}}{\theta_{jN}(\theta_{IN} + \lambda_I - \lambda_j)} \Bigg) \Bigg] + \frac{e^{-(\epsilon_N + \lambda_j)(t-t_{N-1})}}{\theta_{jN}(\theta_{IN} + \lambda_I - \lambda_j)} \Bigg[\frac{S_{jH}(t_{N-1})}{V_1} \left(\frac{Q_1}{V_2}\right)_{N-1} e^{-\sum_{r=1}^{N-2} \tau_r(t_r - t_{r-1})}$$

$$\times \Bigg[\frac{f_p + f_o}{\gamma_{N-1}} \left(e^{-\tau_{N-1}(t_{N-1} - t_{N-2})} - e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})} \right) + \frac{f_e\, e^{-\sum_{r=1}^{N-2} \lambda pr(t_r - t_{r-1})}}{\gamma_{N-1} - \lambda_{pN-1}} \Big(e^{-(\tau_{N-1} + \lambda_{pN-1})(t_{N-1} - t_{N-2})}$$

$$- e^{-\epsilon_{N-1}(t_{N-1} - t_{N-2})} \Big) \Bigg] + \sum_{K=1}^{N-2} \Bigg\{ \left(\frac{Q_1}{V_2}\right)_K \frac{S_{jH}(t_K)}{V_1}\, e^{-\sum_{r=K+1}^{N-1} (\epsilon_r + \lambda_j)(t_r - t_{r-1}) - \sum_{r=1}^{K-1} \tau_r(t_r - t_{r-1})}$$

$$\times \Bigg[\frac{f_p + f_o}{\gamma_K} \left(e^{-\tau_K(t_K - t_{K-1})} - e^{-\epsilon_K(t_K - t_{K-1})} \right) + \frac{f_e\, e^{-\sum_{r=1}^{K-1} \lambda_{pr}(t_r - t_{r-1})}}{\gamma_K - \lambda_{pK}} \Big(e^{-(\tau_K + \lambda_{pK})(t_K - t_{K-1})} - e^{-\epsilon_K (t_K - t_{K-1}}$$

(11a)

$$I'_{42} = E_{3_{jN}} \left(\frac{\chi}{Q_G}\right)_N Q_{E_N} \frac{S_{jH}(t)}{V_1} e^{-\tau_N(t-t_{N-1}) - \sum_{r=1}^{N-1} \tau_r(t_r - t_{r-1})}$$

$$\times \left\{ \frac{f_e e^{-\lambda_{pN}(t-t_{N-1}) - \sum_{r=1}^{N-1} \lambda_{pr}(t_r - t_{r-1})}}{(\delta_{jN} - \lambda_{pN})(\delta_{IN} - \lambda_{pN} + \lambda_I - \lambda_j)} + \frac{f_p + f_o}{\delta_{jN}(\delta_{IN} + \lambda_I - \lambda_j)} \right\}$$

(11b)

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	RUD	1/9/95	MJT	1/9/95	JOB NO. 122-93-01	PAGE 30
					CALC NO. C122-93-01-01	OF 60

$$I'_{43} = \frac{1}{(K_{IN} + \lambda_I)} \sum_{K=1}^{N-1} \left\{ e^{-\sum_{r=K+1}^{N} (K_{Ir}+\lambda_I)(t_r - t_{r-1})} \left[\left(T'_{2_K}(t_K) - e^{-(K_{IK}+\lambda_I)(t_K - t_{K-1})} T'_{2_K}(t_{K-1}) \right) + \left(T'_{3_K}(t_K) - e^{-(K_{IK}+\lambda_I)(t_K - t_{K-1})} T'_{3_K}(t_{K-1}) \right) \right] \right\} \qquad (11c)$$

4.5.7 <u>Control Room Filter Activity</u>

$$\underset{[t_{N-1} \le t \le t_N]}{A_i(t)} = \left[\frac{X}{Q} \right]_N (1-e_{2_N}) e_{3_N} Q_{2_N} Q_{3_N} A_{1_i}(t) + \left[\frac{X}{Q_G} \right]_N e_{3_N} Q_{E_N} Q_{3_N} A_{2_i}(t) + e_{3_N} Q_{6_N} A_{3_i}(t) \qquad (12)$$

A more detailed discussion of the derivation of these equations is presented in Section 5 of Reference 40.

4.6 PADD INPUT DATA

PADD is designed to read virtually all of its input from a series of input files. These files are ASCII text files created by the user using any text editor prior to executing PADD. The user may add comment lines to any of the PADD input files. A comment line is indicated when the first two characters of the line are exclamation points. The input processing part of PADD ignores comment lines. All input in PADD is free format. This means that all data items are to be separated by a comma, and the last data item on a line should <u>not</u> have a comma following it. If data cannot be fit onto one 80 character line, data may be continued on subsequent lines. <u>Character</u> data must also be enclosed in single

quotes. A summary of the required input files is provided below: (*In certain instances where an example of data file name is given, it is merely for demonstration, and should not be construed as the only choice.*)

File Name and Extension	Description
User specified	Input/output file name specification file. This file has only two lines. **Line 1**: Name of the primary input file (with extension, if any) in single quotes. e.g. 'CASE1.IN' **Line 2**: Name of the output file (with extension, if any) in single quotes. e.g. 'CASE1.OUT'
User specified	Primary input file. This file specifies the names of all other input files used with the run and specifies other basic data determining the size and complexity of the run: **Line 1**: Run title (maximum 80 characters) inside single quotes ('_'). **Line 2**: Input file names (maximum 8 characters with extension - see file descriptions below): Name of isotope data file e.g. 'ISOCASE1.DAT' Name of energy group data file e.g. 'ENGCASE1.DAT' Name of breathing rate data file e.g. 'BRECASE1.DAT' Name of time data file e.g. 'TIMCASE1.DAT' Name of filter efficiency data file e.g. 'FILCASE1.DAT' Name of Chi/Q data file e.g. 'CHQCASE1.DAT' **Line 3**: Number of "time steps" (maximum of 30) Number of noble gas isotopes (maximum of 50) Number of halogen isotopes (maximum of 50) Number of dose locations (maximum of 3) Number of gamma energy groups (maximum of 30)

File Name and Extension	Description
	Line 4: Time (in sec) at which plateout is to be "shut off" by code. Fraction of halogens in elemental form. Fraction of halogens in particulate form. Fraction of halogens in organic form. Decontamination factor - at times ≥ that for which plateout is "shut off" (see above), reduction in airborne halogen concentration due to plateout is assumed to be a constant equal to the reciprocal of this factor. **Line 5:** Compartment 1 volume (cubic feet). Compartment 2 volume (cubic feet). Compartment 3 volume (cubic feet).
User specified	Isotope data input file. (*ISOCASE1.DAT*) **Line 1**: Names of noble gas isotopes (maximum 6 characters). e.g. 'KR-81','KR-83M','KR-85'...etc. Names of halogen isotopes (maximum 6 characters). e.g. 'BR-80','BR-80M','BR-82'...etc. *Note that the Noble gas and Halogen listing must begin on a new line.* **Line 2**: Half-lives (sec.) for noble gas isotopes; and Half-lives (sec.) for halogen isotopes. *Note that the Noble gas and Halogen listing must begin on a new line.* **Line 3**: Initial noble gas activities (Curies) released into containment at t=0; and Initial halogen activities (Curies) released into containment at t=0. *Note that the Noble gas and Halogen listing must begin on a new line.* **Line 4**: Mean gamma energy (MeV) emitted by each noble gas isotope (Maximum of 30). **Line 5**: Mean gamma energy (MeV) emitted by each halogen isotope (Maximum of 30). **Line 6**: Activity to beta skin dose rate conversion factors for each noble gas isotope (mrem/yr/pCi/m^3). **Line 7:** Activity to whole body gamma dose rate conversion factor for each noble gas isotope (mrem/yr/uCi/m^3).

File Name and Extension	Description
	Line 8: Inhaled activity dose conversion factor for each Halogen isotope (rem/Ci). **Line 9**: Adult inhaled activity to total body dose conversion factor for each Halogen isotope (rem/Ci). **Line 10**: Child inhaled activity to total body dose conversion factor for each Halogen isotope (rem/Ci). **Line 11**: Infant inhaled activity to total body dose conversion factor for each halogen isotope (rem/Ci).
User specified	Energy group input file (*ENGCASE1.DAT*) The conversion factors identified below are determined through a separate shielding calculation. **Line 1**: Gamma energy endpoints for each of the energy groups (MeV). **Line 2**: Flux to gamma dose conversion factors (rem/MeV/sec). **Line 3**: Flux to gamma dose rate conversion factors (rem/hr/phctons/sec-m^3) for Reactor building shine dose (G2 factors). **Line 4**: Flux to gamma dose conversion factors (rem/hr/photons/sec) for Control room filter shine dose (GA factors). **Line 5**: Flux to gamma dose rate conversion factors (rem/hr/photons/sec-m^3) for Control room gamma cloud dose (GG factors).
User specified	Breathing rate input file (*BRECASE1.DAT*) **Line 1**: Adult breathing rates for each location and time step (M^3/sec). *Note: The breathing rates at each location must begin on a new line.*

File Name and Extension	Description
	Line 2: Child breathing rates for each location and time step (M^3/sec). *Note: The breathing rates at each location must begin on a new line. Additionally, child breathing rates for Containment are not used, but a value must be entered.* **Line 3:** Infant breathing rates for each location and time step (M^3/sec). *Note: The breathing rates at each location must begin on a new line. Additionally, infant breathing rates for Containment are not used, but a value must be entered.*
User specified	Time related input file (*TIMCASE1.DAT*) **Line 1:** Time steps (sec) from 0 to 30 days. (maximum of 30) *The distribution of the time steps is not required to be equally distributed between 0 and 30 days. The number of time steps selected directly influences the accuracy of the numerical integration performed by the program. It is recommended that the minimum time step used is 60 seconds. Additionally, it is important that selection of the time steps accounts for data changes, e.g. assume the adult breathing rate for the reactor building changes from 3.47E-4 m^3/sec to 1.75E-4 m^3/sec at T=8 hrs. To account for this change one should have two time steps at the 8 hr mark, i.e. 8 hrs, and 8 hrs 60 sec. This is repeated for all instances any datum changes.* **Line 2:** Containment by-pass leak rate, "QE" (SCFH). **Line 3:** Containment leak rate, "Q1" (volume %/day). **Line 4:** SGTS flow rate, "Q2" (CFM). **Line 5:** Control room inflow rate, "Q3" (SCFM). **Line 6:** Control room inleakage rate, "Q4" (SCFM). **Line 7:** Control room recirculation flow through filters, "Q6" (SCFM). **Line 8:** Plate-out time constant (sec^{-1}) for each time step. **Line 9:** Control room occupancy factors for each time step.

File Name and Extension	Description
User specified	Filter efficiency input file (*FILCASE1.DAT*) **Line 1**: SGTS filter efficiencies for <u>each</u> Noble gas. *Note: The efficiencies are entered in the same order as were the isotopes in "ISOCASE1.DAT" above. The filter efficiency for each Noble gas isotope must begin on a <u>new</u> line. Additionally, a zero value shall be entered for instances when no efficiency value exists. This feature was retained in the event that a different class of isotope, which <u>can</u> be filtered, is used.* **Line 2**: SGTS filter efficiencies for <u>each</u> Halogen. *Note: The efficiencies are entered in the same order as were the isotopes in "ISOCASE1.DAT" above. The filter efficiency for each Halogen isotope must begin on a <u>new</u> line. Additionally, a zero value shall be entered for instances when no efficiency value exists.* **Line 3**: Control room filter efficiencies for <u>each</u> Noble gas. *Note: The efficiencies are entered in the same order as were the isotopes in "ISOCASE1.DAT" above. The filter efficiency for each Noble gas isotope must begin on a <u>new</u> line. Additionally, a zero value shall be entered for instances when no efficiency value exists.* **Line 4**: Control room filter efficiencies for <u>each</u> Halogens. *Note: The efficiencies are entered in the same order as were the isotopes in "ISOCASE1.DAT" above. The filter efficiency for each Halogen isotope must begin on a <u>new</u> line. Additionally, a zero value shall be entered for instances when no efficiency value exists.*
User specified	CHI/Q factors input file (*CHQCASE1.DAT*) **Line 1**: Elevated release CHI/Q factors for each location; Control room, LPZ followed by EAB. *Note: The CHI/Q factors for each location must begin on a <u>new</u> line.* **Line 2**: Ground release CHI/Q factors for each location; Control room, LPZ followed by EAB. *Note: The CHI/Q factors for each location must begin on a <u>new</u> line.*

Each numbered line of data, as described above, can be separated by a comment line (!!), i.e., the line immediately following the end of "Line 1" and preceding the beginning of "Line 2", can be a comment line.

PADD provides detailed output summarizing the input file names, input data, and all calculated data. Currently, PADD output indicates the program name, version number, run date, and run time on the first page only, and it does not paginate the output. This decision was made to make the program independent of specific printers. A summary of the output tables appearing in a PADD output in the sequence that they appear is provided below:

PADD output (*CASE1.OUT*)

- Summary of Isotope data file (*ISOCASE1.DAT*).
- Summary of Energy group data file (*ENGCASE1.DAT*).
- Summary of Breathing rate data file (*BRECASE1.DAT*).
- Summary of Time data file (*TIMCASE1.DAT*).
- Summary of Filter efficiency data file (*FILCASE1.DAT*).
- Summary of CHI/Q factors data file (*CHQCASE1.DAT*).

The following is calculated output; values are presented for each time step and isotope:

- Containment Noble gas concentrations (Ci/m^3)
- Containment Halogen concentrations (Ci/m^3)
- Reactor building Noble gas concentrations (Ci/m^3)
- Reactor building Halogen concentrations (Ci/m^3)

Control room Noble gas concentrations (Ci/m^3)

Control room Halogen concentrations (Ci/m^3)

Control room emergency filter halogen activities (Ci)

Noble gas activity; elevated release rate from secondary Containment (Ci/sec)

Halogen activity; elevated release rate from secondary Containment (Ci/sec)

Noble gas activity; ground release rate from secondary Containment (Ci/sec)

Halogen activity; ground release from secondary Containment (Ci/sec)

Reactor building gamma shine dose rate to Control room (Rem/hr)

CR emergency filter gamma shine dose rates to Control room (Rem/hr)

Control room emergency filter gamma shine dose to control room (Rem)

Gamma cloud immersion dose rates to Control room (Rem/hr)

Gamma cloud immersion doses to Control room (Rem)

Thyroid dose rates to Control room (Rem/hr)

Thyroid doses to Control room (Rem)

Beta skin dose rates to Control room (Rem/hr)

Beta skin doses to Control room (Rem)

LPZ gamma cloud dose rates (Adult, child, and infant) (Rem/hr)

LPZ gamma cloud doses (adult, child, and [illegible]ant) (Rem)

LPZ adult inhalation total body dose rates (Rem/hr)

LPZ adult inhalation total body doses (Rem)

LPZ adult thyroid dose rates (Rem/hr)

LPZ adult thyroid doses (Rem)

LPZ child inhalation total body dose rates (Rem/hr)

LPZ child inhalation total body doses (Rem)

LPZ child thyroid dose rates (Rem/hr)

LPZ child thyroid doses (Rem)

LPZ infant inhalation total body dose rates (Rem/hr)

LPZ infant inhalation total body doses (Rem)

LPZ infant thyroid dose rates (Rem/hr)

LPZ infant thyroid doses (Rem/hr)

EAB gamma cloud dose rates (adult, child, and infant) (Rem/hr)

EAB gamma cloud doses (adult, child, and infant) (Rem)

EAB adult inhalation total body dose rates (Rem/hr)

EAB adult inhalation total body doses (Rem)

EAB adult thyroid dose rates (Rem/hr)

EAB adult thyroid doses (Rem)

Total dose rates (Rem/hr) and integrated doses (Rem) (dose rates followed by dose)

The above output is presented in tabular form, each column representing a "time step". The final table, "Total dose rates...etc." summarizes the cumulative exposure over the time periods of study for the 13 different dose parameters:

RB Shine
Filter Shine
Control Room Cloud (Whole Body)
Control Room Thyroid
Control Room Beta Skin
LPZ Whole Body
LPZ Adult Thyroid
LPZ Child Whole Body
LPZ Child Thyroid
LPZ Infant Whole Body
LPZ Infant Thyroid
EAB Whole Body
EAB Adult Thyroid

In this study, the focus is on operator thyroid, whole body and beta skin doses.

5.0 DESIGN INPUTS

The design inputs for the CNS post-LOCA control room operator thyroid and whole body dose analysis are:

1. 100% of the noble gases in the reactor and 25% of the iodine instantaneously becomes available for leakage from the primary containment as an aerosol based on TID 14844 in accordance with the CNS USAR.

2. The primary containment volume leaks at a rate of 0.635% weight per day for 30 days in accordance with the CNS USAR.

3. The breathing rate is 347 cc/sec for the first 8 hours, 175 cc/sec for the next 16 hours, and 232 cc/sec thereafter in accordance with Reg. Guide 1.3, Rev. 2.

4. Control room occupancy factors are 1.0 for the first 24 hours, 0.6 for the next 72 hours, and 0.4 thereafter in accordance with Reference 28, Table 1.

5. In accordance with Reg. Guide 1.3, Rev. 2, the source term for this calculation is the instantaneous release to the drywell atmosphere of 100% of the core inventory of noble gases and 25% of the core inventory of Halogens.

6. In accordance with Reg. Guide 1.3, Rev. 2, of the released halogens, 91% are in the elemental form, 5% are in the particulate form, and 4% are in organic form. However, as explained in Section 4.3.1, these factors were adjusted to model flow through the inactive SGTS train. The resulting partitioning factors are 85.86%, 4.06%, and 10.08% for elemental, particulate, and organic halogens, respectively.

7. The overall primary containment integrated leak rate limit defined in CNS Technical Specification 4.7.A.2 is 0.635% of the primary containment volume per day at 58 psig. As described in Section 4.3.2, an additional leakage contribution due to leakage from the ECCS recirculation system is added to this, resulting in a primary containment leak rate of 0.646% per day.

   In CNS Surveillance Procedure 6.3.1.1 (Revision 30), the leak rate of 0.635% per day is calculated to be 316 SCFH. For the case of MSIV leakage, the total limit for containment leakage via the MSIVs is 46 SCFH, as stated in CNS Technical Specification 4.7.A.2. Taking the ratio of 46 SCFH to 316 SCFH and multiplying by 0.635% per day yields a containment leakage rate due to MSIV leakage alone of 0.092% per day.

8. In accordance with Reg. Guide 1.3, Rev. 2, leakage from the primary containment is assumed to pass directly to the SGTS inlet without mixing in the surrounding secondary containment atmosphere and should then be assumed to be released as an elevated plume for those facilities with stacks (i.e., CNS). Therefore, no direct release to the environment from the primary containment was modeled in this analysis.

9. Unless otherwise specified, the design inputs listed in this section were taken from Chapter XIV of the CNS USAR (Reference 10).

10. There is a one minute delay in actuation of the control room emergency filtration unit as described in Section 3.1. This results in one minute of unfiltered inleakage into the control room.

11. To prevent credit for mixing in the reactor building, the reactor building volume specified in the PADD input data is set to 1 ft^3.

12. The turbine building volume used for calculation of the dose contribution from MSIV leakage is 1.8E6 ft^3. This is the volume of the turbine building above the operating floor and the volume of the shielded area on the mezzanine floor less 20% for equipment. [Reference 45]

13. The free air volume of the main condenser is assumed to be 69,940 ft^3. This assumes a 7 foot hotwell water level and is not reduced to account for the volume of the internal structural steel and minor piping. [Reference 46]

14. A release from the turbine building without operation of the HVAC and off-gas systems is assumed to uniformly occur from the entire structure. From Reference 47, the turbine building is 111.5 feet wide, 323.5 feet long, and 104.3 feet high. The distance from the center of the turbine building to the control room ventilation intake is approximately 52 meters. The control room ventilation intake is approximately 53 feet above ground level.

15. Control room filtered flow rates of 375 CFM, 1000 CFM, and 2000 CFM are assumed for this calculation. These flow rates represent the current, planned, and potential future design flow rates, respectively.

The design inputs for the refueling accident analysis include those for the LOCA with the following additional inputs.

1. The fuel assembly is dropped from the maximum height allowed by the fuel handling equipment.

2. The entire amount of potential energy (referenced to the top of the reactor core) is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly in the water above the core and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, the fuel grapple, or the grapple cable breaks, or improper grapplings occur.

3. None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

4. 99% of the halogens released from the rods is retained by the refueling pool water. Since there is no provision in PADD to model this holdup factor, it was taken into account in the PADD input for initial halogen isotope activities.

5. The reactor fuel has an irradiation time of 1000 days at design power up to 24 hours prior to the accident. This assumption results in an equilibrium fission product concentration at the time the reactor is shut down. Longer operating histories do not significantly increase the concentration of the fission products of concern. The 24-hour decay time allows time for the reactor vessel head removed, and the reactor vessel upper internals removed. It is not expected that these evolutions could be accomplished in less than 24 hours.

6. An average of 1.8 percent of the noble gas activity and 0.32 percent of the halogen activity is in the fuel rod plenums and available for release. This assumption is based on fission product release data from defective fuel experiments.

7. Due to the negligible particulate activity available for release in the fuel plenums or from the unmelted fuel, none of the solid fission products are assumed to be released form the fuel.

8. One hundred twenty-five fuel rods are assumed to fail. This was the conclusion of the analysis of mechanical damage to the fuel in Section XIV of the CNS USAR.

9. The fission product activity released to the secondary containment will be in proportion to the removal efficiency of the water in the refueling pool. Since water has a poor affinity for the noble gases they are assumed to be instantaneously released from the pool to the secondary containment.

10. As noted in Section XIV-6.3.4, the removal efficiency of the water for halogens can be defined in terms of the partition factor, for which values between 10^3 and 10^5 have been experimentally determined to be applicable for the conditions under investigation. A partition factor of 10^2 for the halogens has been conservatively assumed for this accident. Thus the computed inhalation exposures will be overestimated by a factor of 10 to 10^3.

11. The conservative assumption is made that instantaneous equilibrium is attained between the refueling pool and secondary containment. In reality, if a true equilibrium is maintained, the effects of plateout or fallout would be compensated for by the evolutions of activity from the refueling pool. Therefore, the effects of plateout and fallout are also neglected.

12. The refueling cavity liquid volume is 3.0×10^4 ft^3 and the effective air volume in the secondary containment is 7.95×10^5 ft^3.

13. The maximum standby gas treatment system (SGTS) flow rate of 1780 CFM is assumed, although the CNS USAR specifies the SGTS removes one secondary containment air volume per day, which is equivalent to 552 CFM.

14. High radiation levels in the reactor building exhaust plenum will isolate the normal reactor building and MG set ventilation systems, and actuate the standby gas treatment system. It is assumed that it takes approximately one minute to isolate the reactor building. During the period, full exhaust flow from the operating reactor building roof.

15. The relative humidity in the secondary containment is 70 percent [Reference 10]. Since the refueling accident does not result in the release of any liquid or vapor to the secondary containment, the normal environmental condition existing prior to the accident will also exist after the accident, except for the addition of the released fission products. The relative humidity in the secondary containment will therefore be considerably below any levels which may be detrimental to the filter media in the Standby Gas Treatment System. However, the SGTS charcoal beds and absolute filter media, as well as the air flowing through the filter system, are heated 10°F above the mixture entering the system, reducing the relative humidity to 70% or less.

6.0 REFERENCES

1. Stone & Webster Letter No. NPPD-31-02, "Review of Control Room Dose Calculations, Cooper Nuclear Station", H.F. Faery to G.S. McClure, December 11, 1989, preliminary.

2. Stone & Webster Calculation 13095.16-PR(D)-002, "Control Room Doses due to Intake and Inleakage of Contaminated Air."

3. Stone & Webster Calculation 13095.16-PR(D)-003, "LOCA Cloud Doses to Control Room."

4. Stone & Webster Calculation 13095.16-PR(D)-005, "Direct Control Room Doses due to 10" Core Spray Pipe Following LOCA."

5. Stone & Webster Calculation 13095.16-PR(D)-007, "Direct Control Room Doses due to Reactor Building Shine."

6. Stone & Webster Calculation 13095.16-UR(D)-001, "Doses in Computer Room due to a LOCA."

7. <u>Control Room Habitability Study</u>, December 11, 1980, Stone & Webster Engineering Company.

8. NPPD Letter CNSS867070, "NPPD Response to NRC Request for Information Concerning Control Room Habitability," J. Pilant to J.E. Gagliardo, January 30, 1986.

9. NRC SER for the FSAR issued February 14, 1973, Sections 6.2.2, 9.4, and 15.2.

10. Cooper Nuclear Station USAR.

11. NPPD Interoffice Memorandum, "Emergency Control Room Ventilation Radiation Monitor," A.D. Sutton to J.R. Hotovy, April 22, 1987.

12. Stone & Webster Calculation 15798.15-UR-001-0, "Integrated 30-Day Post LOCA Control Room Filter Dose."

13. Cooper Nuclear Station Surveillance Procedure 6.3.17.18, "Control Room Envelope Pressurization Test", Revision 0.

14. Cooper Nuclear Station Surveillance Procedure 6.3.17.5, "Emergency Fan Charcoal Leak, Charcoal Sampling, and Fan Capacity Test," Revision 11.

15. GE NEDO-24782, 80NED006, "BWR Owner's Group NUREG-0578 Implementation: Analyses and Positions for Plant-Unique Submittals," August 1980, Chapter 6.

16. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants", Revision 2, March 1978.

17. Burns and Roe Drawing 2037, "Flow Diagram, H&V Standby Gas Treatment and Off Gas Filters," Rev. N38, NPPD Document #454003630.

18. Burns and Roe Drawing 2019, "Flow Diagram, Main Control Room, Cable Room, and Computer Room, Heating, Ventilation, and Air Conditioning," Rev. N17, NPPD Document #454003607.

19. Cooper Nuclear Station USAR, Volume I, Table I-7-1.

20. EDS Nuclear Report 01-0840-1115, Rev. 0, "Cooper Nuclear Station Environmental Effects Due to Pipe Rupture."

21. Standard Review Plan 6.4, Section III.3.d.(2).

22. ERIN Engineering and Research, Inc. Report No. TR122-90-09-01, "Cooper Nuclear Station Control Room Habitability Issues - Independent Review," March 1990.

23. ERIN Engineering and Research, Inc., "Post Accident Design Dose (PADD), Version 1.0Q User Manual," Revision 0, January 1994.

24. NUREG-0588, Rev 1. "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."

25. Regulatory Guide 1.52, Rev. 2, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Absorption Units of Light, Water-Cooled Nuclear Power Plants."

26. NUREG/CR-0009, October 1978.

27. Marks' Standard Handbook for Mechanical Engineers, Eighth Ed., Baumeister, Availone, Baumeister.

28. 13th AEC Air Cleaning Conference, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19, K.G. Murphy and Dr. K.M. Campe.

29. ANS/SD-76/14, July 1976, A Handbook of Radiation Shielding Data, J.C. Courtney, Ed.

30. Nucl. Sci. Eng., Vol. 73 (1980) p 97 "Photon Point Source Buildup Factors for Air, Water, and Iron," Chilton, Eisenhauer, Simmons.

31. Regulatory Guide 1.3, Rev. 2, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors."

32. "Chart of the Nuclides," 11th Ed., Knolls Atomic Power Laboratory.

33. Regulatory Guide 1.25, Rev., "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."

34. NUREG-0800, Rev. 2, Standard Review Plan. Sec 6.4 "Control Room Habitability System."

35. TID-24190, July 1968, Meteorology and Atomic Energy 1968, W.S. AEC

36. NUREG/CR-5055, PNL-6391, May 1988. Atmospheric Diffusion for Control Room Habitability Assessments, J.V. Rausdell, Battelle PNL.

37. U.S. NRC Standard Review Plan 6.4, Rev. 2, July 1981.

38. ICRP (International Commission on Radiological Protection) Publication 30 - iodine inhalation PCFs.

39. Regulatory Guide 1.109, Revision 1 (pages 1.109-21 thru 1.109-22).

40. WPPSS WNP-2 Control Room Dose Calculation NE-02-88-27, Revision 2, April 14, 1992.

41. Southern California Edison Company, Summary of Meeting on June 24, 1986, Regarding Control Room Habitability Requirements, Docket No. 50-206, July 9, 1986.

42. Stone & Webster Calculation 13095.16-PR(D)-001, "Dispersion Coefficients (X/Qs) at the NPPD Cooper Nuclear Station Control Room for Accidental Stack Releases of Radionuclides."

43. Burns and Roe Drawing 2020, "Flow Diagram, Reactor Building H&V," Rev. N25.

44. NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants, Draft for Comment, June 1992.

45. Burns & Roe HVAC Calculation Book 1.

46. Drawing C93877GA, Sheet 1, Rev. N01.

47. Burns & Roe Drawing 4506, "Roof Plan and Details", Rev. N26.

48. Standard Review Plan 15.6.5, Revision 2, July 1981.

49. NPPD Calculation NEDC 94-176, "Radiological Dose Consequences of ECCS Leakage During a LOCA", 8/29/94.

50. CNS STP 94-199, "Control Room Envelope Unfiltered Inleakage Test", 8/5/94.

51. CNS DC 94-102, "Standby Gas Treatment System Cross-tie Valve Modification/Heater Setpoint Change", 1/5/95.

7.0 NOMENCLATURE

The variables used in developing the PADD equations are defined as follows:

a	Operator distance from control room filters
A	A subscript denoting control room emergency bypass filter
A_i	Activity on the control room emergency bypass filters
A_1	Internal surface area of the drywell
B	Buildup factor
B, B,' B," B‴	Constants of Integration
b_1	Number of mean free photon paths in shielding
$C_i\|_o$	Concentration of isotope i at times zero in containment
C_{1i}	Containment concentration of isotope i
C_{2i}	Reactor building concentration of isotope i
C_{3i}	Control room concentration of isotope i
C_{4i}	Air intake concentration of isotope i. As discussed in section 3.4, this can be at any, or all, of a number of points.
C_{5i}	Net concentration of filtered and unfiltered isotope from intakes
d	Control room roof thickness
d_1	Reactor building wall thickness
d_2	Control room wall thickness
D	Total operator gamma dose
D_A	Total gamma dose from control room emergency bypass filter
D_T	Total thyroid dose
D_β	Total beta dose
D_γ	Total dose from control room gammas
D_2	Total gamma dose from reactor building shine
D_1	Total operator dose rate
D_A^1	Dose rate from control room emergency bypass filter
D_T^1	Thyroid dose rate
D_β^1	Beta dose rate
D_γ^1	Dose rate from control room gammas
D_2^1	Dose rate from reactor building shine
E	Gamma energy
E_m	Energy of the m^{th} gamma energy group
E_β	Average beta energy
e_{2i}	The SGTS efficiency for isotope i
e_{3i}	The control room emergency bypass filter efficiency for isotope i

E_{3i}	The expression $\left[\frac{Q_3(1-e_{3i})+Q_4}{V_3}\right]$
f_e	Fraction of halogens which are elemental
f_o	Fraction of halogens which are organic
f_p	Fraction of halogens which are particulate
$F_A(t)$	Fraction of activity which is on control room filters
$F_1(t)$	Fraction of activity which is in containment
$F_2(t)$	Fraction of activity which is in reactor building
$F_3(t)$	Fraction of activity which is in the control room
G_A	Flux to dose conversion factor for control room EFUs
G_γ	Flux to dose conversion factor for control room gammas
G_2	Flux to dose conversion factor for reactor building shine
H	A subscript for halogens (bromine and iodine)
i	Δ subscript for isotope i
I	Operator inhalation rate
j	A subscript for isotope j
L	Containment height divided by $\sqrt{2}$
m	A subscript for gamma energy group m
MAC	Mass attenuation coefficient
n	A subscript for evaluation time n
N	A subscript for noble gases
P_a	Peak post-LOCA drywell absolute pressure
P_b	Standard atmospheric absolute pressure
Q_A	Source strength of a plane source
Q_E	Leak rate from containment directly to the environment
Q_0	Source strength of a point source
Q_1	Leak rate from containment to the reactor building
Q_2	Flow rate through the SGTS
Q_3	"Fresh Air" flow rate through the control room emergency bypass filter
Q_4	"Fresh Air" flow rate into the control room not through the control room emergency bypass filter
Q_5	Control room recirculation rate
Q_6	Recirculation rate through the control room emergency bypass filter
R_T	Adult inhalation thyroid dose conversion factor
R_γ	Gamma flux to dose rate conversion factor
S	Atmospheric stability class 1=Pasquill A, 2=B, ..., 6=F, 7=G
$S_i(t)$	Total activity of isotope i at time t
t	Time from start of LOCA
t_n	Time at the n^{th} evaluation time

t_p	Time at which the plateout term, $e^{\lambda_{pt}}$, equals 1/200
T_a	Peak post-LOCA drywell absolute temperature
T_b	Standard atmosphere absolute temperature
U	Average wind speed resulting in site 5% χ/Q
V_a	Drywell volume at post-LOCA conditions
V_b	Drywell air volume at standard temperature and pressure
V_1	Volume of the primary containment
V_2	Volume of the reactor building
V_3	Volume of the control room, including attached rooms, recirculation ducting and air handling units
X	Distance from activity source to closest remote intake
α_1, α_2, A	Taylor buildup factor equation coefficients
τ	The expression $\dfrac{Q_1 + Q_E}{V_1}$
ϵ	The expression Q_2 / V_2
γ	The expression $\left[\dfrac{Q_2}{V_2} - \dfrac{Q_1 + Q_E}{V_1}\right]$
κ	The expression $\dfrac{Q_3 + Q_4 + e_{3_i} Q_6}{V_3}$
δ	The expression $\left[\dfrac{Q_3 + Q_4 + e_{3_i} Q_6}{V_3} - \dfrac{Q_1 + Q_E}{V_1}\right]$
θ	The expression $\left[\dfrac{Q_3 + Q_c + e_{3_i} Q_6}{V_3} - \dfrac{Q_2}{V_2}\right]$
λ_i	The radioactive decay constant of isotope i
λ_j	The radioactive decay constant of isotope j
λ_p	The plateout decay constant
λ_T	The biological decay constant of iodine isotopes in thyroid
μ	Gamma absorption coefficient
ρ	Density of shield material
σ_y	Lateral diffusing coefficient evaluated for given atmospheric stability and distance between the source and receptor
σ_z	Vertical diffusion coefficient
ϕ_E	Undirected energy flux
χ/Q	Elevated release atmospheric dilution factor "CHI over Q"

χ/Q_G	Ground release atmospheric dilution factor
1	The subscript denoting containment or drywell
2	The subscript denoting the reactor building
3	The subscript denoting the control room
B1A	Adult inhalation dose total body conversion factor (R.G. 1.109)(rem/Ci)
B1C	Child inhalation dose total body conversion factor (R.G. 1.109)(rem/Ci)
B1I	Infant inhalation dose total body conversion factor (R.G. 1.109)(rem/Ci)
BRA	Adult breathing rate ($3.47 \times 10^{-4} m^3/sec$) for control room calcs
BRAA	Adult breathing rate (R.G. 1.109) for LPZ and exclusion area
BRC	Child breathing rate (R.G. 1.109) for LPZ
BRI	infant breathing rate (R.G. 1.109) for LPZ
$EXA_H(t)$	Whole body halogen dose rate to adults in exclusion area
$EXA_N(t)$	Whole body noble gas dose rate to adults in exclusion area
$EXA_{WB}(t)$	Whole body dose rate to adults in exclusion area
EXA_T	Adult thyroid dose rate in the exclusion area
$EXAT_T$	Total adult thyroid dose rate in the exclusion area
DF	Noble beta skin dose conversion factor ($mrem\text{-}m^3$) / (pCi-yr)
$LPZ_{EH}(t)$	Halogens (curies/sec) elevated release offsite
$LPZ_{EN}(t)$	Nobles (curies/sec) elevated release offsite
$LPZ_{GH}(t)$	Halogens (curies/sec) ground release offsite
$LPZ_{GN}(t)$	Nobles (curies/sec) ground release offsite
$LPZ_H(t)$	Whole body halogen dose rate to adults in LPZ
$LPZ_{HC}(t)$	Whole body halogen dose rate to children in LPZ
$LPZ_{HI}(t)$	Whole body halogen dose rate to infants in LPZ
$LPZ_N(t)$	Whole body noble dose rate to adults in LPZ
$LPZ_{NC}(t)$	Whole body noble dose rate to children in LPZ
$LPZ_{NI}(t)$	Whole body noble dose rate to infants in LPZ
$LPZ_{WB}(t)$	Whole body dose rate to adults in emission area
LPZ_T	Adult thyroid dose rate in the LPZ
LPZ_{CT}	Child thyroid dose rate in the LPZ
LPZ_{IT}	Infant thyroid dose rate in the LPZ
$LPZT_T$	Total adult thyroid dose rate in the LPZ
$LPZT_{CT}$	Total child thyroid dose rate in the LPZ
$LPZT_{IT}$	Total infant thyroid dose rate in the LPZ
RH	Halogen conversion (Rem/Ci inhaled) from NRC ICRP30
χ/Q_{LPZE}	Dispersion (sec/m^3) for LPZ from elevated release
χ/Q_{LPZG}	Dispersion (sec/m^3) for LPZ from ground release
χ/Q_{EXE}	Dispersion (sec/m^3) for exclusion area elevated release
χ/Q_{EXG}	Dispersion (sec/m^3) for exclusion area ground release

It is noted that leakage rates, HVAC flow rates, χ/Q factors, and penetration efficiencies are not necessarily constant over time. Consequently, the variables below are defined to allow development of equations which incorporate this potential time dependency.

First, assume that the entire time period over which doses are to be calculated is divided into the following intervals:

INTERVAL n	DEFINITION
1	$t_o \leq t \leq t_1$
2	$t_1 \leq t \leq t_2$
•	•
•	•
•	•
N	$t_{N-1} \leq t \leq t_N$

Second, assume that these intervals are defined such that within each time interval, all HVAC flow rates, leakage rates, χ/Q reactors, and filtration efficiencies are constant.

We can then define the airborne concentration of an isotope i at the end of a given time interval as the Initial Condition for the solution expressing that concentration in the next interval:

$C_{1i}(t_{n-1})$ ≡ airborne concentration if isotope i inside of containment at the end of the (n-1)st time interval

$C_{2i}(t_{n-1})$, $C_{3i}(t_{n-1})$ are similarly defined for the reactor building and control room, respectively

Other variables defined previously (e_{2i}, e_{3i}, E_{3i}, Q_E, Q_1, Q_2, Q_3, Q_4, Q_5, Q_6, δ, ς, θ, χ/Q, χ/Q_G, χ/Q_{LPZE}, χ/Q_{LPZG}, χ/Q_{EXE}, χ/Q_{EXG} τ, κ, ϵ) may appear with a subscript n, denoting the constant value of the variable in the interval n.

8.0 CALCULATIONS

8.1 CALCULATION OF X/Q FACTORS FOR TURBINE BUILDING RELEASE

The model used in this calculation for radionuclide release from the turbine building is the New Building Wake Model from Reference 36. The equation that represents this model, along with a description of how the X/Q values for CNS were developed, is shown below:

$$X/Q = 100(xA)^{-1.2}(U)^{.68}(S)^{.5}$$

where	X/Q	=	relative concentration (s/m^3)
	U	=	windspeed (m/s)
	x	=	distance between release and receptor (m)
	A	=	building area (m^2)
	S	=	numeric stability class identifier (G=7, F=6, etc.)

Given that the turbine building center is roughly southeast of the control room ventilation intake, and considering the relative heights of the source (center of turbine building roof) and receptor (control room ventilation intake), winds from three sectors (SE, ESE and SSE) were considered most likely to impact the control room. Meteorological data for these sectors at a 50 meter elevation [Reference 10, Chapter III] were used in the calculation of X/Q values. The frequency of each windspeed range was normalized with respect to the total frequency of winds from the SE, ESE and SSE.

The selection of X/Qs for the analyses required determining the frequency of exceeding certain values during the four time steps in the analysis. These values are the 95% conditions for 0-8 hrs, 90% conditions for 8-24 hrs, 80% conditions for 1-4 days and 60% conditions for 4-30 days. In these analyses, the representative percentile conditions were

interpreted as percentile X/Qs. This required the calculation of X/Qs for each windspeed and stability class combination. The X/Q values were then sorted in descending order to determine the representative percentile conditions. For example, the 95% X/Q is the X/Q value that is exceeded only 5% of the time. The following table summarizes the representative X/Q values and meteorological conditions determined using the New Building Wake Model.

Time	Percentile	Meteorological Condition	New Building Wake Model X/Q
0-8 hr	95%	D/24 mph	1.52E-03
8-24 hr	90%	E/19 mph	1.45E-03
1-4 days	80%	F/13 mph	1.23E-03
4-30 days	60%	D/13 mph	1.00E-03

In addition to determining the representative windspeed and stability class combinations, a wind direction credit after eight hours was taken in accordance with Reference 28. This credit accounts for the variability of the wind direction over a period of time. For 8 to 24 hours, this factor is 0.88. For 1 to 4 days, the factor is 0.75 and for 4 to 30 days it is 0.5. The net X/Q values incorporating the wind direction factors are shown in the following table.

Time	New Building Wake Model	Correction Factor	Net X/Q
0-8 hr	1.52E-03	1.00	1.52E-03
8-24 hr	1.45E-03	0.88	1.28E-03
1-4 days	1.22E-03	0.75	9.23E-04
4-30 days	1.00E-03	0.50	5.00E-04

The spreadsheets used to develop the raw X/Q (not incorporating the wind direction correction factor) and the meteorological data used in the analysis are shown in Table 8-1. The X/Q values selected are highlighted.

Table 8-1

X/Qs FOR ALL STABILITY CLASS/WINDSPEED COMBINATIONS

NEW BUILDING WAKE MODEL Distance Between Source & Receptor = 52 m Effective Building Area = 1366 m^2				
Stability Class	Windspeed (mph)	X/Q (s/m^3)	Frequency	Cumulative Frequency
7	24	2.01E-03	0.00	0.00
6	24	1.86E-03	0.00	0.00
7	19	1.71E-03	0.00	0.00
5	24	1.70E-03	0.03	0.03
6	19	1.59E-03	0.01	0.04
4	24	1.52E-03	0.01	0.05
5	19	1.45E-03	0.06	0.11
7	13	1.32E-03	0.02	0.13
3	24	1.31E-03	0.00	0.13
4	19	1.29E-03	0.05	0.17
6	13	1.22E-03	0.05	0.22
3	19	1.12E-03	0.00	0.22
5	13	1.12E-03	0.11	0.33
2	24	1.07E-03	0.00	0.33
4	13	1.00E-03	0.11	0.45
7	8	9.51E-04	0.02	0.47
2	19	9.15E-04	0.00	0.47
6	8	8.80E-04	0.03	0.50
3	13	8.66E-04	0.01	0.51
5	8	8.04E-04	0.05	0.57
1	24	7.58E-04	0.00	0.57

W1229301-5630-010695

NEW BUILDING WAKE MODEL Distance Between Source & Receptor = 52 m Effective Building Area = 1366 m^2				
Stability Class	Windspeed (mph)	X/Q (s/m^3)	Frequency	Cumulative Frequency
4	8	7.19E-04	0.09	0.66
2	13	7.07E-04	0.01	0.67
1	19	6.47E-04	0.02	0.69
3	8	6.22E-04	0.01	0.69
7	4	5.93E-04	0.01	0.71
6	4	5.49E-04	0.02	0.72
2	8	5.08E-04	0.01	0.73
5	4	5.02E-04	0.02	0.75
1	13	5.00E-04	0.06	0.81
4	4	4.49E-04	0.04	0.85
3	4	3.88E-04	0.00	0.85
1	8	3.59E-04	0.08	0.94
2	4	3.17E-04	0.00	0.94
7	1	2.31E-04	0.00	0.94
1	4	2.24E-04	0.04	0.99
6	1	2.14E-04	0.00	0.99
5	1	1.95E-04	0.00	0.99
4	1	1.75E-04	0.00	0.99
3	1	1.51E-04	0.00	0.99
2	1	1.24E-04	0.00	0.99
1	1	8.74E-05	0.01	1.00
7	0	0.00E+00	0.00	1.00
6	0	0.00E+00	0.00	1.00
5	0	0.00E+00	0.00	1.00
4	0	0.00E+00	0.00	1.00
3	0	0.00E+00	0.00	1.00
2	0	0.00E+00	0.00	1.00
1	0	0.00E+00	0.00	1.00

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE	
5	RU	1/9/95	MJT	1/9/95	JOB NO. 122-93-01
					CALC NO. C122-93-01-01

ERIN®
ENGINEERING AND RESEARCH, INC.

8.2 RESULTS

As previously stated, all calculations were performed using the PADD computer program. The pertinent input parameters are shown in Table 8-2 for all runs performed. Results for the design basis or "base case" calculations for both the LOCA and refueling accident scenarios are presented in Table 8-3. Additional case study results are also presented in Table 8-3 for sensitivity analyses defined by the NPPD staff. The PADD computer program input and output files for all case studies performed in this calculation are presented in Appendix B.

Table 8-2

CASE STUDY INPUT PARAMETERS

Run #	Leakage Path	Control Room Filtered Intake Flow Rate (CFM)	Control Room Unfiltered Inleakage (CFM)	Primary Containment Leak Rate (vol %/day)	SGTS Flow Rate (CFM)	Primary Containment Volume (ft^3)	Reactor Building Volume (ft^3)	Control Room Volume (ft^3)
LOSS OF COOLANT ACCIDENT CASES								
1a	Primary	375	100	.646	1780	239100	1	141860
1b	MSIV	375	100	.092	.243	239100	1800000	141860
3a	Primary	375	200	.646	1780	239100	1	141860
3b	MSIV	375	200	.092	.243	239100	1800000	141860
5a	Primary	1000	100	.646	1780	239100	1	141860
5b	MSIV	1000	100	.092	.243	239100	1800000	141860
7a	Primary	1000	200	.646	1780	239100	1	141860
7b	MSIV	1000	200	.092	.243	239100	1800000	141860
9a	Primary	2000	100	.646	1780	239100	1	141860
9b	MSIV	2000	100	.092	.243	239100	1800000	141860
11a	Primary	2000	200	.646	1780	239100	1	141860
11b	MSIV	2000	200	.092	.243	239100	1800000	141860
REFUELING ACCIDENT CASES								
13	FHA	375	100	N/A	1780	239100	795000	141860
14	FHA	375	200	N/A	1780	239100	795000	141860
15	FHA	1000	100	N/A	1780	239100	795000	141860
16	FHA	1000	200	N/A	1780	239100	795000	141860

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE		
5	KD	1/9/95	MJT	1/9/95	JOB NO.	122-93-01
					CALC NO.	C122-93-01-01

ERIN®
ENGINEERING AND RESEARCH, INC.

Table 8-3

CASE STUDY RESULTS

Run #	Leakage Path	Operator Thyroid Dose (rem)	Operator Whole Body Dose (rem)	Operator Beta Skin Dose (rem)
LOSS OF COOLANT ACCIDENT CASES				
1a	Primary	4.98	4.18E-02	4.07E-01
1b	MSIV	.39	4.26E-05	1.14E-03
TOTAL		5.37	4.18E-02	4.08E-01
3a	Primary	7.12	4.56E-02	4.31E-01
3b	MSIV	.56	4.76E-05	1.20E-03
TOTAL		7.68	4.56E-02	4.32E-01
5a	Primary	3.66	5.90E-02	5.49E-01
5b	MSIV	.25	4.31E-05	1.20E-03
TOTAL		3.91	5.90E-02	5.50E-01
7a	Primary	4.84	6.09E-02	5.68E-01
7b	MSIV	.35	4.53E-05	1.22E-03
TOTAL		5.19	6.09E-02	5.69E-01
9a	Primary	3.09	7.02E-02	6.75E-01
9b	MSIV	.20	4.33E-05	1.21E-03
TOTAL		3.29	7.03E-02	6.77E-01
11a	Primary	3.78	7.09E-02	6.85E-01
11b	MSIV	.25	4.46E-05	1.22E-03
TOTAL		4.03	7.09E-02	6.86E-01
REFUELING ACCIDENT CASES				
13	FHA	1.90	1.86E-01	1.80
14	FHA	2.40	2.04E-01	1.94
15	FHA	1.69	2.67E-01	2.53
16	FHA	1.95	2.75E-01	2.61

APPENDIX A

CNS MSIV LEAKAGE PATH ASSESSMENT

MSIV LEAKAGE PATH ASSESSMENT
COOPER NUCLEAR STATION
REVISION B

1.0 INTRODUCTION

The Cooper Nuclear Station (CNS) main steam system design includes Main Steam Isolation Valves (MSIVs) on the exit of the primary containment. These valves are arranged in series in four twenty-four inch steam lines for a total of eight (8) valves. These valves function to isolate main steam following a plant trip and, among other things, form a portion of the primary containment boundary for post-LOCA retention of the radionuclide inventory. As part of the CNS post-LOCA control room operator thyroid dose calculation, it has been determined that some contribution of the source term will bypass the primary/secondary containment features via design basis leakage through these valves. A conservative calculational assumption could be made that all of the containment bypass leakage through these pathways is released to the turbine building and is available for transport from this building to the control room. However, it is desired to develop a technical basis for the consideration of retention of much of the released source term in the BOP systems.

The CNS MSIV configuration is not unique among BWRs. In all BWRs the MSIVs are credited to provide a post-accident fission product barrier. The issue of MSIV leakage was reviewed for CNS by the NRC at the time of plant licensing[7] and it was concluded by the NRC that, subject to NPPD performing testing and monitoring of MSIV leakage, the configuration was acceptable when the small probability of the postulated accident conditions concurrent with the failure of the main steamlines outside of containment or the turbine condenser and because of the conservative nature of the staff's analysis of dose consequences. The issue of MSIV leakage and control was later identified as an industry concern by the NRC because of increased MSIV leakage seen at several BWRs. As a result the NRC evaluated the issue, issued Reg. Guide 1.96 and required certain BWRs to install a leakage control system (LCS) for MSIV leakage control. At the time of NRC study of this phenomena CNS was not required to install such a system. This may be due to historical tested MSIV leakage that was less than the Technical Specification value of 11.5 SCFH for a total of 46 SCFH (See History in Attachment A). The NRC and industry studies[2,3] undertaken since that time concerned themselves with plants that were postulated to have leakages in excess of the technical specification limits. The documentation does not provide clear guidance on the consideration of the MSIV leakage in offsite and onsite dose calculations for plants that have no special means to collect or mitigate the MSIV leakage source term.

In previous CNS dose calculations the MSIV leakage was assumed not to add to the source term available for transport. The most recent calculation[4] assumed the MSIV leakage was filtered by the standby gas treatment system (SGTS). The phenomena has been evaluated for other BWRs of a similar design[5,6] and it was determined that the low postulated MSIV leakage

was heldup in the mainsteam systems for a time period that minimized its participation in the offsite doses. In the case of plant Hatch, the NRC concluded that the BOP systems (turbine and condenser) sufficiently hold up the source term, that the allowable MSIV leakage rate was increased to 100 SCFH per valve and 250 SCFH total. It is concluded that such logic also applies to the onsite dose received by the control room operators. However, if such credit is to be taken it must be explicit and the post-accident configuration of the CNS mainsteam system components understood. The evaluation that follows is such an assessment.

2.0 TECHNICAL ASSESSMENT

The discussion above determines that CNS may assume holdup or limitations on the amount of MSIV leakage if the post-accident status of the main steam and turbine control systems is understood. Therefore, an evaluation of leakage pathways and holdup in the turbine system will be undertaken. This evaluation will include consideration of:

1) post-turbine trip status of BOP systems,
2) flow rates as a function of time, and
3) release points.

This review scope is adopted from a the criteria in SRP section 15.6.5, Appendix D, "Radiological Consequences of a Design Basis Loss-of-Coolant-Accident: Leakage from a Main Steam Line Isolation Valve Leakage Control System." CNS does not have a leakage control system (LCS) for MSIV leakage collection, but the assessment methodology is applicable. The following is an assessment for CNS.

2.1 Evaluation of Leakage Potential

Burns and Roe P&ID's[1] and main steam isometrics for CNS were reviewed to identify all potential leakage pathways. the pathways are identified in the table in subsection 2.2. These paths represent a potential "chamber " for containment of the MSIV leakage source term. An effective containment volume for MSIV leakage propagation was calculated based on the Reference 1 drawings, associated isometric drawings and standard pipe size dimensions. This volume is estimated to be approximately 5,000 ft^3.

Considering the maximum MSIV leakage rate of 46 SCFH, it will take approximately 108 hours (4.5 days) for the entire volume to fill with the postulated MSIV leakage. At that time, simplistic consideration of the scenario would predict that continued leakage would be expected to begin to pressurize the system and cause leakage into the turbine building. However, the condenser volume and condensation of steam is expected to have an unquantified, but not insignificant effect on the pressurization of the system. In addition the scenario considered here assumes the MSIVs are leaking at their nominal Technical Specification leakage rate. Lower MSIV leakage would extend the time available before pressurization would begin. It is also assumed

that the off-gas systems for the processing, filtering and releasing of gaseous radwaste through the elevated release point are not available. However, these systems will in all likelihood be available or capable of being restored. Processing of the source term by these systems would minimize the source term as well. This evaluation also neglects the plate-out processes modeled in References 2 and 3 which showed significant source term reduction during the holdup time in the main steam system. Therefore, if containment within this volume can be deterministically shown, it can be postulated that the source term is held up in the BOP systems at CNS.

2.2 Evaluation of Leakage Pathways

Burns and Roe P&ID's for CNS were reviewed to identify all potential leakage pathways for CNS. Each pathway was considered for the following:

1) Destination,
2) Post-LOCA/LOOP (if worst case) configuration,
3) Potential fraction of the source term assigned to pathway.

The assessment assumes that a flow path destination of the condenser will contain the source term and preclude leakage. The assessment combines LOCA/loss of offsite power (LOOP), but does not evaluate the seismic ruggedness of these systems as was done for Hatch[6]. This is acceptable for the reasons discussed by the NRC in the CNS SER[7]. The following is a table of results of that assessment.

CNS POST-ACCIDENT MSIV LEAKAGE PATHWAYS

No.	Line Number	Description	Post-Trip Condition	Disposition
1	3"-MS-1S	3" main steam line drain via MO-78 to the condenser.	MS-MOV-MO78 is closed in both the normal operating configuration and emergency post-trip configuration.	The path is normally isolated. If open the final destination is the condenser.
2	30"-MS-1	Main steam lines to the high pressure turbine via SV1BV & SV2BV	Turbine stop valves MS-HOV-SV1 and SV2 and turbine control valves MS-HOV-GV1, GV2, GV3, and GV4 close on a turbine trip.	This pathway is isolated as part of post-trip actuations. Destination is the high pressure turbine chest. In the event of incomplete isolation, seal leakage would be expected.
3	18"-MS-1	Turbine bypass to condenser 1A & 1B via turbine bypass valves HO-BV1(2,3).	Turbine bypass valves MS-HOV-BV1, BV2 and BV3 close in the absence of steam pressure.	Post-trip condition is isolated. In the event the valves remain open, the final destination is the condenser.
4	2"-MS-1	Main steam to the steam jet air ejectors via BV-1A(B) and PCV-77A(B) to 3"-BS-2	Pressure control valves MS-AOV-PCV77A and B do not close unless there is a loss of power or air.	The event postulated is a LOCA combined with a LOOP which would cause a loss of power. However, the worst case is that post-trip availability of air and power, resulting in no isolation and a potential pathway for release. Final destination is the condenser and /or the low pressure turbine.
5	5"-MS-1	Turbine Gland and Condenser sealing steam via PCV-68A and MO-IMV3(BMV3).	The turbine gland seal system check valve MS-CV-18CV is closed in the absence of steam pressure.	Final destination is the condenser. This holds up the source term.

W1229301-5630-042294

No.	Line Number	Description	Post-Trip Condition	Disposition
6	2"-MS-1	Turbine bypass line steam trap 15.	Trap bypass valve MS-AOV-195AV is closed during all conditions and fails closed. The trap allows line to drain condensate.	Final destination is the condenser. This holds up the source term.
7	2"-MS-1	Turbine bypass strainers drain.	Valves do not isolate line: however, strainers dP would not allow flow if no steam pressure present.	Final destination is the condenser. This holds up the source term.

3.0 CONCLUSION

The above assessment evaluates all flow paths out of the CNS BOP systems that could result in release of the post-accident source term leaking through the MSIVs. A LOCA/loss of offsite power (LOOP) is considered and consequential effects considered, as applicable. The evaluation concludes that containment within the BOP systems is deterministically shown and the migration of the MSIV leakage source term within those systems is a slowly acting phenomena that will not challenge the integrity of the system. Therefore, it can be postulated that the source term is held up in the BOP systems at CNS.

4.0 REFERENCES

1.0 CNS Drawings:

P&IDs

Burns and Roe Drawing 2002, Sheet 1 of 3, Rev. N29, "Flow Diagram Main Exhaust & Auxiliary Steam Systems".

Burns and Roe Drawing 2002, Sheet 2 of 3, Rev. N25, "Flow Diagram Main Exhaust & Auxiliary Steam Systems".

Burns and Roe Drawing 2002, Sheet 3 of 3, Rev. N07, "Flow Diagram Main Exhaust & Auxiliary Steam Systems".

Burns and Roe Drawing 2041, Rev. N53, "Flow Diagram Reactor Bldg. Main Steam System".

Isometrics

JELCO Incorporated Drawing 2809-5, Rev. N01, "BS-2 Pipe to Jet".

JELCO Incorporated Drawing 2841-1, Rev. 10, "Main Steam Supply From Turbine to 36" Header, Turbine Building".

JELCO Incorporated Drawing 2841-2, Rev. N03, "MS-1 Lines From MS-1 36" Header, Turbine Building".

JELCO Incorporated Drawing 2842-1, Rev. N01, "MS-1 Lines From Turbine Bypass Valve to Condensers".

2.0 NUREG-1169, "Resolution of Generic Issue C-8, An Evaluation of Boiling Water Reactor Main Steam Isolation Valve Leakage and the Effectiveness of Leakage Treatment Methods", dated August 1986.

3.0 Battelle Memos, Del Lessor to Jim Jamison/Dennis Strenge, Subject: MSIV Project, dated September 6, 1985 (Revised Copy), September 16, 1985, September 24, 1985, and September 30, 1985 (Ret# 17839 0561).

4.0 Stone & Webster Calculation 13095.16-PR(D)-002, "Control Room Doses due to Intake and Inleakage of Contaminated Air" dated December 11, 1980.

5.0 Duane Arnold Energy Center-1 (DAEC) UFSAR, Section 6.7, "Main Steam Line Isolation Valve Leakage Control System", Revision 2 - 6/84.

6.0 Letter, USNRC to Georgia Power Company, Kahtan N. Jabbour (USNRC) to J. T. Bechham, Jr., Subject: Issuance of Amendment - Edwin I. Hatch Nuclear Plant, Unit 2, dated March 17, 1994.

7.0 CNS Safety Evaluation Report, Section 6.2.2 Isolation Systems, dated February 14, 1973.

ERIN CALC. NO.
C122-93-01-01

ATTACHMENT A
TO APPENDIX A
PAGE 1 OF 4

CNS

LOCAL LEAK RATE TEST HISTORY

PENETRATION: X-7A

DESCRIPTION: Main Steam Isolation Valves - Line A

ALLOWABLE LEAKAGE (SCFH): 11.5/Valve

CIC: MS-AOV-AO80A
MS-AOV-AO86A

AS FOUND LEAKAGE			AS LEFT LEAKAGE	
YEAR	TOTAL	PEN.	TOTAL	PEN.
1973	4.31	2.16	4.31	2.16
1975	81.63	36.08	1.77	0.89
1976	6.65	3.33	6.65	3.33
1977	5.32	2.66	5.32	2.66
1978	14.81	4.09	14.81	4.09
1979	9.30	4.65	9.30	4.65
1980	7.98	3.99	7.98	3.99
1981	6.21	3.11	6.21	3.11
1982	1.52	0.76	1.52	0.76
1983	13.42	6.71	0.42	0.21
1984	11.15	5.58	11.15	5.58
1986	2.34	1.17	2.34	1.17
1988	10.86	5.43	10.86	5.43
1989	28.15	6.68	2.37	1.19
1990	0.153	0.0765	0.153	0.0765
1991	1.66	0.83	1.66	0.83
1993	1.32	0.66	7.56	3.78

NOTE: TOTAL IS MAX PATH LEAKAGE
PEN IS MIN PATH LEAKAGE

5838

ERIN CALC. No.
C122-93-01-01
ATTACHMENT A
TO APPENDIX A
PAGE 2 OF 4

CNS

LOCAL LEAK RATE TEST HISTORY

PENETRATION: X-7B

DESCRIPTION: Main Steam Isolation Valves - Line B

ALLOWABLE LEAKAGE (SCFH): 11.5 / Valve

CIC: MS-AOV-AO80B
MS-AOV-AO86B

AS FOUND LEAKAGE			AS LEFT LEAKAGE	
YEAR	TOTAL	PEN.	TOTAL	PEN.
1973	2.43	1.22	2.43	1.22
1975	13.82	4.78	13.82	4.78
1976	11.49	5.75	11.49	5.75
1977	12.89	6.45	10.34	5.17
1978	2.29	1.15	2.29	1.15
1979	6.60	3.30	6.60	3.30
1980	8.50	4.25	8.50	4.25
1981	8.45	4.23	8.45	4.23
1982	12.94	3.03	12.94	3.03
1983	8.37	4.19	8.37	4.19
1984	9.37	4.69	3.66	1.83
1986	2.45	1.23	2.45	1.23
1988	6.54	3.27	6.54	3.27
1989	5.75	2.88	0.84	0.42
1990	0.229	.01145	0.229	0.1145
1991	0.90	0.45	0.90	0.45
1993	2.92	1.46	4.27	2.135

EKIN CALC. NO.
C122-93-01-01

CNS

LOCAL LEAK RATE TEST HISTORY

ATTACHMENT A
TO APPENDIX A
PAGE 3 of 4

PENETRATION: X-7C

DESCRIPTION: Main Steam Isolation Valves - Line C

ALLOWABLE LEAKAGE (SCFH): 11.5 / Valve

CIC: MS-AOV-AO80C
MS-AOV-AO86C

AS FOUND LEAKAGE			AS LEFT LEAKAGE	
YEAR	TOTAL	PEN.	TOTAL	PEN.
1973	0.93	0.47	0.93	0.47
1975	8.63	4.32	8.63	4.32
1976	10.72	5.36	10.72	5.36
1977	8.24	4.12	8.24	4.12
1978	6.44	3.22	6.44	3.22
1979	4.50	2.25	4.50	2.25
1980	9.89	4.95	9.89	4.95
1981	8.24	4.12	8.24	4.12
1982	12.09	3.36	12.09	3.66
1983	12.93	6.47	1.60	0.80
1984	4.95	2.48	4.95	2.48
1986	5.82	2.91	5.82	2.91
1988	8.42	4.21	8.42	4.21
1989	4.81	2.41	1.59	0.80
1990	1.18	0.59	1.18	0.59
1991	4.16	2.08	4.16	2.08
1993	2.47	1.235	6.61	3.305

ERIN CALC No.
C122-93-01-01

ATTACHMENT A
TO APPENDIX A
PAGE 4 OF 4

CNS

LOCAL LEAK RATE TEST HISTORY

PENETRATION: X-7D

DESCRIPTION: Main Steam Isolation Valves - Line D

ALLOWABLE LEAKAGE (SCFH): 11.5 / Valve

CIC: MS-AOV-AO80D
MS-AOV-AO86D

AS FOUND LEAKAGE			AS LEFT LEAKAGE	
YEAR	TOTAL	PEN.	TOTAL	PEN.
1973	8.96	4.48	8.96	4.48
1975	14.38	2.89	14.38	2.89
1976	11.88	5.36	11.88	5.36
1977	9.23	4.62	9.23	4.62
1978	4.44	2.22	4.44	2.22
1979	8.90	4.45	8.90	4.45
1980	14.20	6.20	6.20	3.10
1981	15.97	7.54	15.97	7.54
1982	12.19	5.94	12.19	5.94
1983	20.20	10.10	0.0	0.0
1984	6.14	3.07	6.14	3.07
1986	2.38	1.19	2.38	1.19
1988	6.77	3.39	6.77	3.39
1989	3.08	1.54	0.0	0.0
1990	6.69	3.345	6.69	3.345
1991	0.48	0.24	0.48	0.24
1993	2.78	1.39	5.04	2.52

APPENDIX B

PADD OUTPUT FROM CASE STUDIES

The following PADD output reports are included in this appendix:

CASE NUMBER	RUN NAME	RUN DATE/TIME	TOTAL PAGES
1	RUN1A	1-06-95 / 11:24:43	98
2	RUN1B	1-09-95 / 12:19:52	98
3	RUN3A	1-06-9[illegible] / 11:46:46	98
4	RUN3B	1-09-95 / 12:42:45	98
5	RUN5A	1-06-95 / 12:46:31	98
6	RUN5B	1-09-95 / 12:50:59	98
7	RUN7A	1-06-95 / 13:35:12	98
8	RUN7B	1-09-95 / 12:59:13	98
9	RUN9A	1-06-95 / 13:51:36	98
10	RUN9B	1-09-95 / 13:07:28	98
11	RUN11A	1-06-95 / 14:08:16	98
12	RUN11B	1-09-95 / 13:15:38	98
13	RUN13	1-06-95 / 20:31:45	98
14	RUN14	1-06-95 / 20:36:34	98
15	RUN15	1-06-95 / 20:40:42	98
16	RUN16	1-06-95 / 20:45:51	98

Although additional doses (i.e., filter shine, reactor building shine, exclusion area doses, and low population zone doses) are listed in the PADD output, these doses use data that has not been verified and, thus, are not valid.

W1229301-5630-010695

REV.	BY	DATE	CHECKED	DATE
5	RD	1/9/95	MJT	1/9/95

JOB NO. 122-93-01

CALC NO. C122-93-01-01

ERIN®
ENGINEERING AND RESEARCH, INC.