GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

MFN-019-83
JNF-007-83

March 9, 1983

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. D. G. Eisenhut, Director
Division of Licensing

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT
(GESSAR II) DOCKET NO. 50-447

CHAPTER 15 RADIOLOGICAL DOSE CALCULATIONS WITH
POOL SCRUBBING CREDIT

Reference: Glenn G. Sherwood to D. G. Eisenhut, "Revision of
Chapter 15 Radiological Dose Calculations,"
January 25, 1983.

The purpose of this letter is to document agreements made with the Staff regarding credit for suppression pool scrubbing in Chapter 15 radiological dose calculations.

On January 13, 1983, GE personnel met with Mr. L. G. Hulman and other members of the Staff to discuss credit for pool scrubbing for GESSAR II Chapter 15 accident evaluations. In that meeting, the Staff indicated that if GE provided the analysis of reduced radiological doses resulting from pool scrubbing, the Staff would recognize the potential decontamination capability attributed to the pool. This would be reflected in the GESSAR II SER in March, 1983. In addition the Staff would provide a detailed review of the submittal and would prepare a supplement to the SER which would include pool scrubbing credit. It was indicated such a supplement could be issued by September 1983. In accordance with this approach, GE submitted the required information on the GESSAR II docket (reference letter).

As discussed in this meeting the January 25 submittal culminates several technical meetings with the Staff on the subject of suppression pool decontamination factors.

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GENERAL ELECTRIC

If there are any questions on the information provided herein, or in the referenced letter, please contact J. N. Fox of my staff at (408) 925-5039.

Very truly yours,

Willa for

Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

GGS:pes/112R

Attachment

cc: F. J. Miraglia, NRC
D. C. Scaletti, NRC
L. G. Hulman, NRC
C. O. Thomas, NRC
R. M. Ketcheil, GE (Washington Liaison Office)
L. S. Gifford, GE (Bethesda Liaison Office)

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 682, (408) 925-5040

January 25, 1983

MFN-011-83

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Mr. D. G. Eisenhut, Director
Division of Licensing

Gentlemen:

SUBJECT: IN THE MATTER OF 238 NUCLEAR ISLAND
GENERAL ELECTRIC STANDARD SAFETY ANALYSIS REPORT
DOCKET NO. 50-447

REVISION OF CHAPTER 15 RADIOLOGICAL DOSE CALCULATIONS

Attached please find a revision of the Chapter 15 radiological dose calculations for the Inside Containment Loss of Coolant Accident. Also included are the affected pages of Chapter 6.

These calulations have been revised to account for the capability of the suppression pool to retain particulate fission products. As demonstrated by General Electric's suppression pool scrubbing test program, and documented in Section 15D. 2, the suppression pool provides an extremely effective fission product retention mechanism. General Electric personnel have been working with your staff during 1982 to facilitate the review of these test results and their application to GESSAR's Chapter 15 analysis.

This revision demonstrates the capability of the suppression pool to significantly reduce offsite radiological doses for Design Basis Accident conditions. Following your preliminary review we will schedule the submittal of an amendment to implement this revision.

Very truly yours,

Glenn G. Sherwood, Manager
Nuclear Safety and Licensing Operation

Attachments

cc: F.J. Miraglia, w/o att.
D.C. Scaletti
L.G. Hulman
C.O. Thomas, w/o att.
L.S. Gifford, w/o att.

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6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment (Continued)

outdoor air are mixed and drawn through filters, a cooling coil and zone electric reheating coils.

There are two intakes, a normal intake located on the roof of the Control Building, and an alternate intake on the opposite side of the Nuclear Island at the end of the Auxiliary Building. Radiation monitoring sensors located in each duct warn the operating personnel (by means of readouts and alarms in the main control room) of the presence of airborne contamination. Also, the signal automatically closes down the normal air intake dampers and starts up the reduced flow (2000 cfm) air intake. This alternate service air, which is classified as makeup air, is routed through the HEPA and charcoal filtering system for cleanup before being used for pressurization.

The control room must remain habitable during emergency conditions. In order to make this possible, potential sources of danger such as steam lines, pressure vessels, CO_2 fire fighting containers, etc. are located outside of the control room and removed from the compartments containing Control Building life support systems.

A tabulation of moving components in the Control Building HVAC System, along with the respective failure mode and effects, is shown in Table 6.4-1.

All dampers except the mixing dampers in the air conditioning units are of the two position (open or closed) type.

6.4.2.5 Shielding Design

6.4.2.5.1 Design Basis

The Control Building shielding design is based upon adequately protecting against the radiation resulting from ~~an incident or incidents leading to~~ a POSTULATED LOCA. The dose rates received under normal operating conditions of the reactor are not determining factors in any of the walls sized in this specification. Under normal operating conditions, a dose rate of less than 1 mRem/hr is anticipated in the area surrounding the Control Building. Assuming an average gamma energy of 1.25 MeV and 2-ft shielding walls, this will yield a dose rate of less than 0.01 mRem/hr within the Control Building, which is well within the acceptable limit.

Radioactivity released by an inadequate response to LOCA can result in four different activity distributions, or sources, that can affect Control Building personnel whole body doses.

(1) The fission products held in the containment "shine" on the Control Building. (Those remaining in the reactor vessel, however, contribute negligibly to this effect.)

(2) The fission products which are released from the SGTS stack form a cloud in which the Control Building is enveloped (Figure 6.4-5).

(3) Some of the fission products released from the containment will be taken into the Control Building via the building ventilation system air intakes. The majority of the iodine taken in will be absorbed on a charcoal bed, which will then become a concentrated source within the building. Also, solid daughters of noble gases

6.4.2.5.1 Design Basis (Continued)

collect on the filters. Personnel on the control room level, as well as the equipment room and HVAC room levels, will be shielded from this source.

(4) Fission products that pass through and evolve from the filters become a source of radiation exposure to control building personnel. This source determines a portion of the whole body dose, as well as the entirety of the thyroid and beta skin doses. See Subsection 15.6.5 for these dose analyses.

The DBA analysis is structured on the conservative NRC assumptions. The QUANTITY OF ~~percentages of 105% rated power~~ fission productS ~~equilibrium inventories~~ released FROM THE REACTOR VESSEL TO THE CONTAINMENT ~~from the reactor vessel and available for release from the containment are given below:~~ IS PRESENTED IN SUBSECTION 15.6.5

~~Fission Product~~	~~Released from Reactor Vessel~~	~~Available for Release from Containment~~
~~Noble Gases~~	~~100%~~	~~100%~~
~~Halogens~~	~~50%~~	~~25%~~
~~Solids~~	~~1%~~	~~Negligible~~

The containment leak rate assumed for the design analyses is 1.0% of the containment volume per day. Radioactive decay during transport through the containment is taken into account. The leaked radioactivity goes into the SECONDARY CONTAINMENT ~~shield building annulus,~~ and then to the SGTS, from which it is vented to the atmosphere. Mixing is assumed to occur in half of the shield building annulus free volume. The SGTS charcoal filter is assumed to be 99% efficient for filtering ORGANIC AND PARTICULATE FORMS OF ~~radio~~iodine~~s~~, and none of the vented gas is assumed to bypass the filter.

6.4.2.5.2 Source Terms

Containment WETWELL sources "shining" on the Control Building AND POTENTIALLY AVAILABLE FOR RELEASE TO THE ENVIRONMENT are listed in Table 15.6.~~5-2~~-8. Source terms for the cloud and filter are consistent with the activity releases of Table 15.6.~~5-3~~-11. Concentration of each isotope is calculated as the product of the release rate (Ci/sec) times the appropriate relative concentration, or χ/Q (sec/~~square~~ meter^3). ~~These values, derived from "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19", by Kampe and Murphy, are presented below:~~

~~Time Period~~	~~(χ/Q (sec/m^3) LOCA in Same Unit~~
~~0-8 hrs~~	~~0.0015~~
~~8-24 hrs~~	~~0.0015~~
~~1-4 days~~	~~0.0006~~
~~4-183 days~~	~~0.0005~~

For the cloud source term, no credit was taken for decay between the release point and the cloud location. Buildup and decay of radiohalogens and solids on the filter was appropriately accounted for.

6.4.2.5.3 Results

~~Calculated whole body doses to control room personnel are given below for an assumed six-month post-LOCA period. One hundred percent occupancy factor is assumed.~~
WHOLE BODY GAMMA DOSES CALCULATED BY CONSERVATIVELY ASSUMING 100% OCCUPANCY FOR A SIX-MONTH POST-LOCA PERIOD ARE GIVEN BELOW FOR THE CONTAINMENT, CLOUD AND FILTER SHINE CONTRIBUTIONS.

6.4.2.5.3 **Results (Continued)**

Dose Component	Location (Dose in Rem)	
	Control Room	Equipment Room
Containment Shine	0.0054	0.0063
Cloud	0.19	0.20
Charcoal Filter	0.0012	0.078
	0.20	0.28

WHOLE BODY, THYROID, AND SKIN DOSES RESULTING FROM THE AIRBORNE ACTIVITY WITHIN THE CONTRC ROOM ARE EVALUATED IN SECTION 15.6.5 AND ARE SHOWN TO BE WELL WITHIN THE LIMITS OF 10 CFR 50. THE COMBINED WHOLE BODY DOSE RESULTING FROM INCLUSION OF THE SMALL ~~That the above doses are much less than the 10CFR50 guidelines,~~ INCREMENTAL DOSES FROM THE SOURCES CONSIDERED ABOVE IS STILL WELL WITHIN THE LIMIT. ~~implies the acceptability of full~~ THUS, occupancy of these areas for the DURATIO OF THE post-LOCA period IS ASSURED. Occupancy will, however, have to be restricted somewhat in the chiller rooms at El (+)28'-6" during the first day post-LOCA, due to somewhat high dose rates (up to 0.5 Rem/hr attributable to radioactive gases in the intake duct. However, safety-related equipment redundancy obviates the need for full occupancy. After the first day, full occupancy in these areas would result in less than 5 Rem.

Concrete shielding thickness effecting the above doses are seen in plan and elevation in Figures 6.4-1 through 6.4-4. Penetrations and resultant streaming through the adjacent walls of the Auxiliary and Control Buildings are not considered to be significant for the following reasons. The penetrations are all relatively small (cable trays less than 24 in, pipes less than 6 in.) and not radially aligned with the containment. Also, the overall contribution of the containment source component is quite small; thus, an increase due to streaming would not be a significant overall increase. Figure 6.4-6 shows a cross section of the Division 1 and 2 HVAC air intake. This is the only significant penetration of the Control Building external shielding. Examination of this figure shows no credible streaming for the external cloud source. Finally, arrangement of the filter cubicle precludes

6.4.2.5.3 Results (Continued)

door streaming. There are no floor penetrations of any consequence, and wall penetrations are kept well away from the source region.

6.4.3 System Operational Procedures

During normal operation, the control room group operates with mixed recirculated and fresh air, which pressurizes the subject spaces. Emergency conditions such as LOCA or high radiation cause automatic change over to reduced outside air and charcoal filtering of all outside air to effectively isolate operating personnel from the environment and from airborne contamination. Protection from direct radiation is discussed in Subsection 6.4.2.5; isolation can be complete, even to food and water (Subsection 6.4.4).

Detection of radioactivity is instrumented, and changeover to reduced circulation and charcoal filtering is automatic. Redundancy of instrumentation and air handling systems ensures against system failure due to single component failure.

The above operational description is brief. For a more detailed description of normal and emergency operation of the control room habitability systems, see Subsections 9.4.1, 9.5.1, 9.5.3, 12.3.4, 6.5.1, 7.3.1.1.17, and Chapter 8.

6.4.4 Design Evaluations

6.4.4.1 Radiological Protection

Assumptions used in the generation of post-LOCA radiation source terms are described fully in Subsections ~~6.2.4.5 and~~ 15.6.5.

15.6.5.3.3 Results

Results of this event are given in detail in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured. Continued long-term core cooling is demonstrated. Radiological input is minimized and within limits. Continued operator control and surveillance is examined and guaranteed.

15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and NOT EXCEED LEVEL C ACCEPTANCE CRITERIA ~~experience normal stresses~~ after the instantaneous rupture of the largest single primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Sections 3.8.2.3, 3.6, and 6.2 for details and results of the analyses).

15.6.5.5 Radiological Consequences

Two separate radiological analyses are provided for this accident:

(1) The first is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet

15.6.5.5 Radiological Consequences (Continued)

10CFR100 guidelines. This analysis is referred to as the "design basis analysis".

(2) The second is based on assumptions considered to provide a MORE realistic estimate of radiological consequences. This analysis is referred to as the "realistic analysis", ALTHOUGH MANY CONSERVATIVE ASSUMPTIONS STILL REMAIN.

A schematic of the transport pathway is shown in Figure 15.6-2.

Additional parameters and information for specific design basis accidents are provided in Subsection 19.3.15.1. 15.1

15.6.5.5.1 Design Basis Analysis

INSERT A

~~The methods, assumptions and conditions used to evaluate this accident are in accordance with those guidelines set forth in Regulatory Guides 1.3 and 1.7. The specific models, assumptions and computer code used to evaluate this event based on the above criteria are presented in Reference 2. Specific values of parameters used in this evaluation are presented in Table 15.6-7.~~

15.6.5.5.1.1 Fission Product Release from Fuel

INSERT B

It is assumed that 100% of the noble gases and 50% of the iodine are released from an equilibrium core operating at a power level of 3651 MWt for 1000 days prior to the accident. While not specifically stated in Regulatory Guide 1.3, the assumed release of 100% of the core noble gas activity and 50% of the iodine activity implies fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for the evaluation of this accident. Of this release, 100% of the noble gases and 50% of the iodine become airborne. The remaining 50% of the iodine is removed by plate-out and condensation; therefore, it is not available for airborne release to the environment. The activity airborne in the containment is presented in Table 15.6-8.

6

15.6.5.5.1.2 Fission Product Transport to the Environment

The transport pathway consists of leakage from the containment to the secondary containment-like structures by several different mechanisms and discharge to the environment through the Standby Gas Treatment System (SGTS):

6

(1) Containment leakage - The design basis leak rate of the primary containment and its penetrations (excluding the main steamlines) is 1.0%/day for the duration of the accident. All of this leakage is to the secondary containment and from there to the environment via a 99% SGTS. Credit is taken for mixing and holdup within the secondary containment. The Shield Building exhaust rate, leakage rate, and mixing ratio are given on Tables 15.6-9 and 15.6-10.

(2) Leakage from engineered safety feature (ESF) components outside primary containment.

(~~3~~ 4) Hydrogen purge - In the event of failure of the Hydrogen Recombiner System, purging of the containment may be necessary to control hydrogen concentration inside the primary containment. The earliest this purge may be utilized is one hour after the accident AT A rate of 100 scfm minimum. The purge would be processed by SGTS prior to release to the environment. SINCE THE HYDROGEN CONTROL EQUIPMENT IS ESF EQUIPMENT, IT IS NOT ASSUMED TO FAIL IN EVALUATING THE POTENTIAL RADIOLOGICAL EXPOSURES ASSOCIATED WITH THIS ACCIDENT.

Fission product release to the environment based on the above assumptions is given in Table 15.6-11.

15.6.5.5.1.3 Results

~~The calculated exposures for the design basis analysis are presented in Table 15.6-12 AND are well within the guidelines of 10CFR100.~~

INSERT C

INSERT A (page 15.6-15)

The methods, assumptions and conditions used to evaluate the potential radiological exposure to onsite and offsite personnel are consistent with the guidelines set forth in Regulatory Guide 1.3 except as noted below.

The guidance of Regulatory Guide 1.3 has been supplemented using information from: 1) NUREG-0772 to account for the expected chemical forms of the fission products; and 2) Section 15 D.2 to account for the retention of particulate fission products in the suppression pool. The specific models, assumptions and computer code used to evaluate this event based on the above criteria are presented in Reference 2. Specific values of parameters used in this evaluation are presented in Table 15.6-7.

INSERT B (p 15.6-15)

It is assumed that 100% of the noble gases and 50% of the iodine are released to the drywell from an equilibrium core operat~~ing~~ed at a power level of 3651 MWt for 1000 days prior to the accident. These assumed releases imply that severely degraded ECCS performance has resulted in fuel damage approaching melt conditions. Even though this condition is inconsistent with operation of the ECCS system (Section 6.3), it is assumed applicable for this evaluation of containment system effectiveness.

Any iodine which is released from the reactor vessel would exist predominantly in a particulate chemical form. This analysis assumes that chemical forms of iodine released to the drywell are distributed in accordance with NUREG-0772, with 99.97% being in the particulate form and 0.03% being organic iodine. Of the particulate iodine released, 50% is assumed to be removed by plateout and condensation and therefore is not available for potential release to the environment.

15.6.5.5.1.2 FISSION PRODUCT TRANSPORT TO THE ENVIRONMENT

The Mark III containment is designed with the drywell and suppression pool totally enclosed within the containment wetwell. In this configuration, any fission products released to the drywell, or discharged through the safety/relief valves, must pass through the suppression pool before they can reach the containment wetwell airspace. Once in the primary containment airspace, the transport pathway consists of limited leakage from the primary containment to the secondary containment by several different mechanisms and discharge to the environment through the Standby Gas Treatment System (SGTS). Consistent with the SGTS design capability and in accordance with R.G. 1.52 and the BWR/6 Standard Technical Specifications, it is assumed that the SGTS has an iodine removal efficiency of 99%.

~~These~~ Transport pathways and ~~any~~ associated retention mechanisms are identified below:

INSERT B CONTINUED

1) <u>Suppression Pool</u> - All fission products discharged from the drywell or safety/relief valves enter the suppression pool 7 to 8 feet under the pool surface. As described in Section 15 D.2, General Electric has performed tests which demonstrate that the suppression pool will act as an extremely effective retention device for particulate fission products discharged into the pool under these conditions. Based upon the testing and modeling documented in Section 15 D.2, a suppression pool scrubbing decontamination factor (DF) equal to 10,000 was applied to reduce the activity of particulate iodine reaching the primary containment airspace. All noble gases and organic forms of iodine were assumed to pass through the pool without retention (i.e., DF = 1). The resulting activity airborne in the containment wetwell airspace due to noble gases and iodine is presented in Table 15.6-8.

2) <u>Primary Containment Leakage</u> - The design basis leak rate of the primary containment and its penetrations is 1.0%/day for the duration of the accident. All of this leakage is to the secondary containment and from there to the environment via the SGTS. Parameters applicable to TRANSPORT FROM the primary and secondary containments are given on Tables 15.6-9 and 15.6-10. ~~and the activity airborne in the primary containment is presented in Table 15.6-8~~.

3) <u>Engineered Safety Feature Leakage</u> - Engineered safety feature (ESF) components located outside the primary containment contribute insignificant leakage to the secondary containment.

INSERT C (p15.6-16)

The results of the design basis analysis are presented in Table 15.6-12. The calculated exposures at the Exclusion Area and Low Population Zone are well within the guidelines of 10CFR100.

These calculated results are believed to be significantly greater than the maximum potential dose due to the consérvative assumptions used in the design basis analysis. For example, the calculated exposures would be significantly reduced if a more realistic time dependent release of fission products from the fuel were assumed. Further, other fission product retention mechanisms are present, in addition to the suppression pool, which would act to limit the release of fission products to the environment. These additional retention mechanisms include agglomeration and settling of particulate forms plus surface deposition and absorption in the water vapor which would exist in the containment air space following a LOCA. An additional retention mechanism exists as a result of the containment sprays.

A sensitivity study has been performed to determine to what extent these results would change if the fuel release source term model recommended by NUREG-0772 were used instead of the Regulatory Guide 1.3 source term. If the NUREG-0772 model for a fully melted core were applied, the fuel release would change from 100% noble gas and 50% iodine to that specified in Table F.3-1 of Section 15D.3. Since the noble gas release remains at a maximum value of 100% and the suppression pool and SGTS would reduce any increase in the particulate release by a factor of 10^6, the effect on the offsite WHOLE BODY dose would be negligible. Only the thyroid dose would be calculated to change, with an increase from 0.05 rem to 0.1 rem in the Exclusion Area and 0.1 rem to 0.2 rem in the low population zone. Thus, the results would still remain well within the guidelines of 10CFR100.

15.6.5.5.2 Realistic Analysis

INSERT D

The realistic analysis is based on a realistic but still conservative assessment of this accident. ~~The specific models, assumptions and the program used for computer evaluation are described in Reference 3. Specific values of parameters used in the evaluation are presented in Table 15.6-7.~~

15.6.5.5.2.1 Fission Product Release from Fuel

Since this accident does not result in any fuel damage, the only activity released to the drywell is that activity contained in the reactor coolant plus any additional activity which may be released as a consequence of reactor scram and vessel depressurization.

While there are various activation and corrosion products contained in the reactor coolant, the products of primary importance are the iodine isotopes I-131 to I-135. The coolant concentration for these isotopes is:

I-131	~~2.03 E 2.02 μCi/gm~~ 0.02 μCi/gm
I-132	~~2.59 E 1.26 μCi/gm~~ 0.26 μCi/gm
I-133	~~1.49 E 1.15 μCi/gm~~ 0.15 μCi/gm
I-134	~~4.75 E 1.48 μCi/gm~~ 0.48 μCi/gm
I-135	~~2.39 E 1.24 μCi/gm~~ 0.24 μCi/gm

~~Considering that approximately 40% of the released liquid flashes to steam, it is conservatively assumed that 40% of the released iodine activity is airborne initially. However, as a result of plate-out and condensation effects, only 50% of the activity initially airborne remains available for release to the environment.~~

15.6.5.5.2.1 Fission Product Release from Fuel (Continued)

As a consequence of reactor scram and depressurization, additional iodine activity is released from those rods which experienced cladding perforation during normal operation. Measurements performed (Reference ~~8~~ 3) at operating BWRs during reactor shutdown have been used to develop an analytical model for the prediction of iodine and noble gas spiking as a consequence of reactor scram and vessel depressurization. Based on the 95th percentile (i.e., only 5% of the time will the release be greater) probability, the I-131 release is calculated to be 2.14 Ci/bundle and Xe-133 to be 11.55 Ci/bundle. Other iodine and noble gas isotopes are determined in accordance with their cumulative fission yields and are tabulated in Table 15.6-13.

~~While~~ SINCE no measurements have been obtained during a pressure transient as rapid as the LOCA, it is difficult to predict the actual release rate from the fuel as a consequence of iodine spiking. Therefore, it is arbitrarily assumed that 100% of the spiking source term ~~is~~ HAS BEEN released TO THE DRYWELL BEFORE THE BLOWDOWN TRANSIENT IS OVER ~~during the time period that 40% of the discharge coolant is flashing to steam~~. THE IODINE FRACTIONS INVOLVED IN THIS RELEASE ARE CONSISTENT WITH THOSE SPECIFIED IN NUREG 0772 (i.e., 99.97% IN PARTICULATE FORM AND 0.03% AS ORGANIC IODINE). It is also assumed that plate-out and condensation remove~~s~~ 50% of the airborne PARTICULATE iodine activity. ~~The total activity airborne in the containment is presented in Table 15.6-14~~.

15.6.5.5.2.2 Fission Product Transport to the Environment

INSERT → E

~~The leak rate from the primary containment to the secondary containment is 1.0%/day, where 100% mixing is assumed to occur. Release from the secondary containment to the environment via a 99.9% iodine efficient SGTS is presented in Table 15.6-15. The integrated isotopic activity released to the environment is presented in Table 15.6-15.~~

INSERT D (p. 15.6-17)

The chemical forms of iodine used in this realistic analysis are consistent with those specified in NUREG-0772. As in the case of the design basis analysis, appropriate credit is given for suppression pool scrubbing. Specific values of the parameters used in the evaluation are presented in Table 15.6-7. The specific models, assumptions and the program used for computer evaluation are described in Reference 2.

INSERT E (p 15.6-18)

The activity released in this analysis is directed to the drywell and from there into the suppression pool. Any fission products not retained in the suppression pool due to scrubbing are assumed to be distributed in the wetwell airspace. This transport process is discussed in detail in Section 15.6.5.5.1. As in the case of the design basis analysis, credit was taken for a decontamination factor of 10,000 for particulate iodine due to the suppression pool. The activity airborne in the containment wetwell airspace is presented in Table 15.6-14.

Parameters applicable to the primary and secondary containmentSare given in Table 15.6-10. The integrated isotopic activity release to the environment via the SGTS is presented in Table 15.6-15. The SGTS has a 99% ORGANIC iodine removal efficiency AND A 99.9% PARTICULATE IODINE REMOVAL EFFICIENCY.

15.6.5.5.2.3 Results

AS SHOWN IN TABLE 15.6-16,
The calculated radiological exposures for this event are ~~presented in Table 15.6-16 and as shown are~~ a small fraction of 10CFR100.

15.6.5.5.3 Control Room

INSERT F

A dose analysis has been performed to demonstrate that the ventilation system satisfies the NRC radiation guidelines. The results of the analysis show that the ventilation system design does satisfy their guideline. A schematic of the control room intake vents is shown in Figure 15.6-3.

The doses received during a 30-day period after a loss-of-coolant accident are:

6

	Dose (Rem)	U.S. NRC Limit (Rem)
Whole Body	2.56	5
Thyroid	29.4	30
Beta	53.8	75

A factor of 1/4 was taken into account for a dual inlet with manual override capabilities. The methods used to calculate these doses are presented in Reference 5. A complete list of assumptions and input data follows:

(1) Source Terms

The source terms used in this analysis are consistent with R.G. 1.3 (i.e., 25% halogens and 100% noble gases airborne in the containment) and were presented in Table 15.6-8.

15.6.5.5.3 Control Room (Continued)

ADDITIONAL ASSUMPTIONS ON VENTILATION PARAMETERS AND METEOROLOGICAL DATA ARE GIVEN BELOW

Ventilation Parameter (SEE FIGURE 15.6-3 FOR VENTILATION INTAKE SCHEMATIC)

Inlet air flow	
filtered	0.944 m^3/sec
unfiltered	0.0014 m^3/sec
INLET Filter IODINE efficiency	99%
Control Room Volume	1.102 E+4 m^3
Occupancy factors	
0-2 hrs	1.0
2-8 hrs	1.0
8-24 hrs	1.0
1-4 days	0.6
4-30 days	0.4

Meteorology Data

X/Q Values	sec/m^3
0-2 hrs	8.0 E-3
2-8 hrs	3.2 ~~1.6~~ E-3
8-24 hrs	2.8 ~~1.4~~ E-3
24 hr-4 days	2.2 ~~1.1~~ E-3
4-30 days	2.2 ~~1.1~~ E-3

15.6.6 Feedwater Line Break - Outside Containment

In order to evaluate large liquid process line pipe breaks outside containment, the failure of a feedwater line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the feedwater line, representing the largest liquid line outside the containment, provides the envelope

INSERT F (p. 15.6-19)

The control room and its associated ventilation system ~~has~~ HAVE been designed with the objective of continuous occupancy following a LOCA. An analysis has therefore been performed to demonstrate that the ventilation system satisfies the NRC's control room habitability guidelines relative to radiation exposure. The potential doses to control room personnel during a 30-day period after a LOCA are shown below. These are based on: 1) the source terms, iodine fractions and scrubbing discussed in Section 15.6.5.5.1 ; 2) a factor of 0.25 to take credit for a dual inlet with manual override capabilities; and 3) the calculation methods presented in Reference 4.

	Dose (Rem)	U.S. NRC Limit (Rem)
Whole Body	2.9*	5
Thyroid	0.02	30
Beta	36	75

* A SMALL INCREMENT TO THE WHOLE BODY DOSE RESULTS FROM GAMMA SOURCES EXTERNAL TO THE CONTROL ROOM. THIS EFFECT IS EVALUATED IN SECTION 6.4.2.5.

15.6.7 References

1. F. J. Moody, "Maximum Two-Phase Vessel Blowdown from Pipes", ASME Paper Number 65-WA/HT-1, March 15, 1965.

2. ~~P. P. Stancavage and E. J. Morgan, "Conservative Radiological Accident Evaluation - The CONAC01 Code", March 1976 (NEDO-21143)~~.

~~3. D. Nguyen, "Realistic Accident Analysis - The RELAC Code", October 1977 (NEDO-21142)~~.

~~4~~3 F. J. ~~P~~Brutschy, G. R. Hills, N. R. Horton, A. J. Levine, "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup", August 1972 (NEDO-10585).

~~5. D. G. Weiss and V. O. Nguyen, "Control Room Accident Exposure Evaluation - CRDS Program", January 1979 (NEDO-23909)~~

#2 D. NGUYEN, et al, "RADIOLOGICAL ACCIDENT EVALUATION - THE CONAC 03 CODE", DECEMBER 1981 (NEDO-21143-1).

#4 D. NGUYEN, et al, "CONTROL ROOM ACCIDENT EXPOSURE EVALUATION - CRDOS PROGRAM", FEBRUARY 1981 (NEDO-23909A)

Table 15.6-6

STEAMLINE BREAK ACCIDENT (REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	1.1E-2	5.2E-1
Low Population Zone	5.4E-3	2.6E-1

Table 15.6-7

LOSS-OF-COOLANT ACCIDENT - PARAMETERS TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3651 MWt	3651 MWt
B. Burnup	NA	NA
C. Fuel damage	100%	0
D. Release of activity by nuclide	100% NOBLE GAS 50% IODINE	100% NOBLE GAS 50% IODINE
E. Iodine fractions		
(1) Organic	~~0~~ 0.0003	~~0~~ 0.0003
(2) Elemental	~~x~~ 0	~~x~~ 0
(3) Particulate	~~0~~ 0.9997	~~0~~ 0.9997
F. Reactor coolant activity before the accident	NA	15.6.5.5.2.1
II. Data and assumptions used to estimate activity released		
A. Primary containment leak rate ~~(%/day)~~	TABLE 15.6-10 ~~1.0~~	TABLE 15.6-10 ~~1.0~~
B. Secondary containment leak rate ~~(%/day)~~	~~0.2 hrs 319.3~~ TABLE 15.6-10 ~~>10 hrs 43.9~~	~~123~~ TABLE 15.6-10 ~~123~~
C.		
D ~~C~~. Valve movement times	NA	NA
E ~~D~~. Adsorption and filtration efficiencies (%)		
(1) Organic iodine	~~NA~~ 99	~~NA~~ 99
(2) Elemental iodine	~~99~~ NA	~~99.9~~ NA
(3) Particulate iodine	~~NA~~ 99	~~NA~~ 99.9
(4) Particulate fission products	~~NA~~ 99	~~NA~~ 99.9
F ~~E~~. Recirculation system parameters		
(1) Flow rate (CFM)	5000	5000
(2) Mixing efficiency	50	100
(3) Filter efficiency	NA	NA
G ~~F~~. Containment spray parameters (flow rate, drop size, etc.)	NA	NA
H ~~G~~. Containment Volumes	NA	NA
I ~~H~~. All other pertinent data and assumptions	None	None

C. SUPPRESSION POOL SCRUBBING DECONTAMINATION FACTOR

(1) ORGANIC IODINE	1	1
(2) PARTICULATE IODINE	10,000	10,000

Table 15.6-7 (Continued)

	Design Basis Assumptions	Realistic Basis Assumptions
III. Dispersion Rate		
A. Boundary and LPZ distance (m)	*	*
B. χ/Q's for time intervals of		
(1) 0-2 hr - SB/LPS	2.0E-3/1.0E-3	2.0E-3/1.0E-3
	1.0E-3	1.0E-3
(2) 2-8 hr - LPZ	3.8E-4	3.8E-4
(3) 8-24 hr - LPZ	1.0E-4	1.0E-4
(4) 1-4 days - LPZ	3.4E-5	3.4E-5
(5) 4-30 days - LPZ	7.5E-6	7.5E-6
IV. Dose Data		
A. Method of dose calculation	Reference 2	Reference 2
B. Dose conversion assumptions	Reference 2	Reference 2
C. Peak activity concentrations in containment WETWELL AIRSPACE	Table 15.6-8	Table 15.6-14
D. Doses	Table 15.6-12	Table 15.6-16

*Applicant to Supply

REPLACE WITH NEW TABLE 15.6-8

Table 15.6-8

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.1E 07	2.0E 07	1.9E 07	1.5E 07	1.2E 06
I132	3.5E 07	3.0E 07	2.6E 07	1.9E 07	1.0E 07	3.1E 06	9.2E 05	2.4E 04	7.3E-06	0.
I133	3.3E 07	3.2E 07	3.1E 07	3.0E 07	2.8E 07	2.5E 07	2.2E 07	1.4E 07	1.3E 06	9.3E-04
I134	5.5E 07	3.7E 07	2.5E 07	1.1E 07	2.3E 06	9.8E 04	4.1E 03	3.0E-01	0.	0.
I135	4.6E 07	4.3E 07	4.1E 07	3.7E 07	3.0E 07	2.0E 07	1.3E 07	3.6E 06	1.8E 03	0.
Total I	1.9E 08	1.6E 08	1.4E 08	1.2E 08	9.2E 07	6.8E 07	5.6E 07	3.7E 07	1.6E 07	1.2E 06
Kr83m	9.5E 06	8.0E 06	6.6E 06	4.5E 06	2.1E 06	4.8E 05	1.1E 05	1.2E 03	2.2E-09	0.
Kr85m	2.3E 07	2.1E 07	2.0E 07	1.7E 07	1.2E 07	6.6E 06	3.6E 06	5.5E 05	7.6E 00	0.
Kr85	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.9E 05	5.8E 05	5.8E 05	5.6E 05	4.3E 05
Kr87	4.7E 07	3.6E 07	2.7E 07	1.6E 07	5.3E 06	5.9E 05	6.6E 04	9.2E 01	6.5E-16	0.
Kr88	6.7E 07	5.9E 07	5.2E 07	4.1E 07	2.5E 07	9.2E 06	3.4E 06	1.7E 05	2.9E-03	0.
Kr89	6.7E 07	1.2E 05	1.6E 02	3.0E-04	1.1E-15	0.	0.	0.	0.	0.
Xe131m	5.7E 05	5.7E 05	5.7E 05	5.7E 05	5.7E 05	5.6E 05	5.5E 05	5.4E 05	4.4E 05	7.5E 04
Xe133m	2.3E 07	2.3E 07	2.3E 07	2.2E 07	2.2E 07	2.1E 07	2.0E 07	1.7E 07	6.4E 06	1.5E 03
Xe133	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.3E 08	1.2E 08	1.2E 08	7.6E 07	1.9E 06
Xe135m	3.6E 07	9.8E 06	2.5E 06	1.7E 05	7.2E 02	1.4E-02	2.6E-07	0.	0.	0.
Xe135	2.4E 07	2.3E 07	2.3E 07	2.1E 07	1.8E 07	1.3E 07	9.8E 06	3.9E 06	1.6E 04	0.
Xe137	1.5E 08	7.8E 05	3.4E 03	6.8E-02	2.6E-11	0.	0.	0.	0.	0.
Xe138	1.6E 08	3.9E 07	9.0E 06	4.8E 05	1.4E 03	1.1E-02	8.9E-08	0.	0.	0.
Total NG	7.4E 08	3.5E 08	3.0E 08	2.5E 08	2.2E 08	1.8E 08	1.6E 08	1.4E 08	8.3E 07	2.4E 06

Table 15.6-8

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN PRIMARY CONTAINMENT WETWELL (Ci)

ISOTOPE	1-MIN	10-MIN	1-HOUR	2-HOUR	4-HOUR	8-HOUR	12-HR	1-DAY	4-DAY	30-DAY
I-131	1.49E 04	1.49E 04	1.49E 04	1.48E 04	1.47E 04	1.45E 04	1.42E 04	1.36E 04	1.02E 04	8.32E 02
I-132	2.44E 04	2.34E 04	1.82E 04	1.34E 04	7.35E 03	2.20E 03	6.57E 02	1.76E 01	6.44E-09	1.00E-20
I-133	2.28E 04	2.26E 04	2.20E 04	2.13E 04	1.99E 04	1.74E 04	1.52E 04	1.01E 04	8.93E 02	6.41E-07
I-134	3.82E 04	3.39E 04	1.75E 04	7.95E 03	1.64E 03	6.91E 01	2.92E 00	2.20E-04	1.00E-20	1.00E-20
I-135	3.20E 04	3.15E 04	2.86E 04	2.59E 04	2.10E 04	1.38E 04	9.05E 03	2.56E 03	1.31E 00	1.00E-20
TOTAL I	1.32E 05	1.26E 05	1.01E 05	8.34E 04	6.46E 04	4.79E 04	3.91E 04	2.63E 04	1.10E 04	8.32E 02
KR-83M	9.53E 06	9.01E 06	6.57E 06	4.49E 06	2.11E 06	4.62E 05	1.01E 05	1.07E 03	1.50E-09	1.00E-20
KR-85M	2.28E 07	2.23E 07	1.96E 07	1.68E 07	1.23E 07	6.61E 06	3.56E 06	5.53E 05	7.78E 00	1.00E-20
KR-85	5.87E 05	5.87E 05	5.87E 05	5.87E 05	5.86E 05	5.85E 05	5.85E 05	5.82E 05	5.64E 05	4.33E 05
KR-87	4.67E 07	4.30E 07	2.73E 07	1.58E 07	5.32E 06	6.00E 05	6.77E 04	9.73E 01	8.53E-16	1.00E-20
KR-88	6.65E 07	6.41E 07	5.23E 07	4.10E 07	2.51E 07	9.45E 06	3.55E 06	1.89E 05	4.28E-03	1.00E-20
KR-89	6.75E 07	9.43E 06	1.69E 02	3.38E-04	1.36E-15	1.00E-20	1.00E-20	1.00E-20	1.00E-20	1.00E-20
XE131M	5.73E 05	5.73E 05	5.72E 05	5.70E 05	5.67E 05	5.60E 05	5.54E 05	5.35E 05	4.36E 05	7.40E 04
XE133M	2.30E 07	2.29E 07	2.27E 07	2.24E 07	2.18E 07	2.06E 07	1.95E 07	1.66E 07	6.22E 06	1.27E 03
XE-133	1.34E 08	1.33E 08	1.33E 08	1.32E 08	1.30E 08	1.27E 08	1.24E 08	1.16E 08	7.56E 07	1.87E 06
XE135M	3.63E 07	2.44E 07	2.66E 06	1.87E 05	9.17E 02	2.21E-02	5.34E-07	1.00E-20	1.00E-20	1.00E-20
XE-135	2.42E 07	2.40E 07	2.25E 07	2.08E 07	1.79E 07	1.31E 07	9.67E 06	3.85E 06	1.53E 04	1.00E-20
XE-137	1.46E 08	2.87E 07	3.37E 03	6.49E-02	2.41E-11	1.00E-20	1.00E-20	1.00E-20	1.00E-20	1.00E-20
XE-138	1.60E 08	1.03E 08	8.92E 06	4.74E 05	1.34E 03	1.06E-02	8.45E-08	1.00E-20	1.00E-20	1.00E-20
TOTAL NG	7.37E 08	4.85E 08	2.96E 08	2.55E 08	2.16E 08	1.79E 08	1.62E 08	1.38E 08	8.28E 07	2.38E 06

Table 15.6-9

SHIELD BUILDING EXHAUST RATE

Time (hr)	Average Exhaust Flow Rate to SGTS (SCFM)
0 - 2	480
2 - 10	90
>10	66

Table 15.6-10

LEAKAGE RATES AND MIXING RATIO

Parameter	Numerical Value	
	Design Basis	Realistic
A. Primary to Secondary Containment (%/day)		
0 - 2 hr	0.832	~~1.0~~ 0.832
2 - 10 hr	0.903	~~1.0~~ 0.903
>10 hr	0.908	~~1.0~~ 0.908
B. Primary Containment Leakage to SGTS (%/day)		
0 - 2 hr	0.168	~~NA~~ 0.168
2 - 10 hr	0.097	~~NA~~ 0.097
>10 hr	0.092	~~NA~~ 0.092
C. Secondary Containment Leakage to SGTS (%/day)		
0 - 2 hr	319.3	~~123~~ 159.6
2 - 10 hr	59.9	~~123~~ 29.9
>10 hr	43.6	~~123~~ 22.0
D. Mixing Efficiency (%)		
Primary Containment	100	100
Shield Building Annulus	50	100

REPLACE WITH NEW TABLE 15.6-11

Table 15.6-11

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.5E-01	8.7E 00	2.0E 01	4.8E 01	7.5E 01	1.5E 02	2.4E 02	5.7E 02	3.9E 03	1.8E 04
I132	4.1E-01	1.3E 01	2.8E 01	5.7E 01	7.5E 01	9.6E 01	1.0E 02	1.1E 02	1.1E 02	1.1E 02
I133	3.8E-01	1.3E 01	2.9E 01	7.0E 01	1.1E 02	2.0E 02	3.0E 02	6.0E 02	1.6E 03	1.7E 03
I134	6.4E-01	1.8E 01	3.4E 01	5.7E 01	6.4E 01	6.6E 01	6.6E 01	6.6E 01	6.6E 01	6.6E 01
I135	5.4E-01	1.8E 01	4.0E 01	9.1E 01	1.3E 02	2.2E 02	2.9E 02	4.1E 02	4.8E 02	4.8E 02
Total I	2.2E 00	7.1E 01	1.5E 02	3.2E 02	4.6E 02	7.4E 02	1.0E 03	1.8E 03	6.2E 03	2.0E 04
Kr83m	1.1E 01	3.5E 02	7.3E 02	1.4E 03	1.9E 03	2.2E 03	2.3E 03	2.4E 03	2.4E 03	2.4E 03
Kr85m	2.7E 01	8.9E 02	1.9E 03	4.3E 03	6.2E 03	9.4E 03	1.2E 04	1.4E 04	1.5E 04	1.5E 04
Kr85	6.9E-01	2.4E 01	5.4E 01	1.3E 02	2.1E 02	4.2E 02	6.7E 02	1.7E 03	1.3E 04	1.4E 05
Kr87	5.5E 01	1.7E 03	3.3E 03	6.0E 03	7.2E 03	7.9E 03	8.0E 03	8.1E 03	8.1E 03	8.1E 03
Kr88	7.8E 01	2.5E 03	5.4E 03	1.1E 04	1.6E 04	2.1E 04	2.4E 04	2.5E 04	2.5E 04	2.5E 04
Kr89	8.8E 01	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02	4.7E 02
Xe131m	6.7E-01	2.3E 01	5.3E 01	1.3E 02	2.0E 02	4.0E 02	6.5E 02	1.6E 03	1.1E 04	6.4E 04
Xe133m	2.7E 01	9.3E 02	2.1E 03	5.1E 03	7.9E 03	1.6E 04	2.4E 04	5.5E 04	2.6E 05	4.5E 05
Xe133	1.6E 02	5.4E 03	1.2E 04	3.0E 04	4.7E 04	9.3E 04	1.5E 05	3.5E 05	2.2E 06	7.3E 06
Xe135m	4.4E 01	8.2E 02	1.1E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03
Xe135	2.8E 01	9.7E 02	2.1E 03	5.0E 03	7.5E 03	1.3E 04	1.8E 04	2.8E 04	4.0E 04	4.0E 04
Xe137	1.9E 02	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03	1.2E 03
Xe138	1.9E 02	3.5E 03	4.5E 03	4.8E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03	4.9E 03
Total NG	9.0E 02	1.9E 04	3.5E 04	7.1E 04	1.0E 05	1.7E 05	2.4E 05	4.9E 05	2.6E 05	8.1E 06

Table 15.6-11

LOSS-OF-COOLANT ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASE TO ENVIRONMENT (Ci)

ISOTOPE	1-MIN	10-MIN	1-HOUR	2-HOUR	4-HOUR	8-HOUR	12-HR	1-DAY	4-DAY	30-DAY
I-131	1.75E-04	1.84E-03	1.37E-02	3.34E-02	5.[illegible]4E-02	1.05E-01	1.67E-01	4.01E-01	2.75E 00	1.23E 01
I-132	2.87E-04	2.95E-03	1.93E-02	3.99E-02	5.28E-02	6.75E-02	7.31E-02	7.58E-02	7.59E-02	7.59E-02
I-133	2.67E-04	2.79E-03	2.06E-02	4.93E-02	7.58E-02	1.42E-01	2.13E-01	4.21E-01	1.10E 00	1.20E 00
I-134	4.51E-04	4.45E-03	2.39E-02	3.96E-02	4.46E-02	4.62E-02	4.63E-02	4.63E-02	4.63E-02	4.63E-02
I-135	3.75E-04	3.90E-03	2.79E-02	6.40E-02	9.41E-02	1.55E-01	2.04E-01	2.87E-01	3.37E-01	3.37E-01
TOTAL I	1.56E-03	1.59E-02	1.05E-01	2.26E-01	3.20E-01	5.16E-01	7.03E-01	1.23E 00	4.31E 00	1.40E 01
KR-83M	1.12E 01	1.14E 02	7.24E 02	1.44E 03	1.84E 03	2.21E 03	2.32E 03	2.35E 03	2.35E 03	2.35E 03
KR-85M	2.68E 01	2.78E 02	1.94E 03	4.34E 03	6.18E 03	9.42E 03	1.16E 04	1.41E 04	1.48E 04	1.48E 04
KR-85	6.89E-01	7.23E 00	6.41E 01	1.32E 02	2.08E 02	4.18E 02	6.72E 02	1.66E 03	1.33E 04	1.43E 05
KR-87	5.50E 01	5.54E 02	3.27E 03	6.03E 03	7.24E 03	7.97E 03	8.08E 03	8.09E 03	8.09E 03	8.09E 03
KR-88	7.82E 01	8.05E 02	5.41E 03	1.15E 04	1.57E 04	2.12E 04	2.39E 04	2.56E 04	2.58E 04	2.58E 04
KR-89	8.85E 01	4.12E 02	4.70E 02	4.70E 02	4.70E 02	4.70E 02	4.70E 02	4.70E 02	4.70E 02	4.70E 02
XE131M	6.72E-01	7.05E 00	5.27E 01	1.28E 02	2.02E 02	4.04E 02	6.46E 02	1.56E 03	1.13E 04	6.40E 04
XE133M	2.70E 01	2.83E 02	2.10E 03	5.09E 03	7.93E 03	1.55E 04	2.42E 04	5.45E 04	2.58E 05	4.38E 05
XE-133	1.57E 02	1.64E 03	1.23E 04	2.98E 04	4.67E 04	9.30E 04	1.48E 05	3.50E 05	2.22E 06	7.29E 06
XE135M	4.36E 01	3.76E 02	1.11E 03	1.22E 03	1.23E 03	1.23E 03	1.23E 03	1.23E 03	1.23E 03	1.23E 03
XE-135	2.85E 01	2.97E 02	2.15E 03	5.01E 03	7.49E 03	1.30E 04	1.79E 04	2.82E 04	3.91E 04	3.91E 04
XE-137	1.88E 02	9.82E 02	1.20E 03	1.20E 03	1.20E 03	1.20E 03	1.20E 03	1.20E 03	1.20E 03	1.20E 03
XE-138	1.92E 02	1.63E 03	4.48E 03	4.84E 03	4.85E 03	4.85E 03	4.85E 03	4.85E 03	4.85E 03	4.85E 03
TOTAL NG	8.97E 02	7.38E 03	3.52E 04	7.12E 04	1.01E 05	1.71E 05	2.45E 05	4.93E 05	2.60E 06	8.03E 06

Table 15.6-12

LOSS-OF-COOLANT ACCIDENT (DESIGN BASE ANALYSIS) RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	19.1	~~66.7~~ 0.05
Low Population Zone	14.1	~~14.9~~ 0.11

Table 15.6-13

ISOTOPIC SPIKING ACTIVITY

Isotope Name	The 95th Cumulative Probability Spiking Activity (Ci/bundle)
I131	2.14
I132	3.21
I133	5.03
I134	5.44
I135	4.79
Kr83m	9.04-1*
Kr85m	2.23+0
Kr85	4.90-1
Kr87	4.33+0
Kr88	6.12+0
Kr89	7.96+0
Xe131m	6.60-2
Xe133m	3.26-1
Xe133	1.16+1
Xe135m	1.80+0
Xe135	1.10+1
Xe137	1.05+1
Xe138	1.06+1

*9.04-1 = 9.04 x 10-1

REPLACE WITH NEW TABLE 15.6-14

Table 15.6-14

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE CONTAINMENT (Ci)

Isotope	1 min	1 hr	2 hr	8 hr	1 day	3 day	26 day
I131	8.59E 01	8.55E 01	8.52E 01	8.33E 01	7.84E 01	5.96E 01	5.64E 00
I132	1.41E 02	1.05E 02	7.73E 01	1.25E 01	9.69E-02	2.99E-11	0.
I133	2.08E 02	2.01E 02	1.94E 02	1.59E 02	9.28E 01	8.26E 00	6.91E-09
I134	2.39E 02	1.10E 02	4.97E 01	4.29E-01	1.34E-06	0.	0.
I135	2.03E 02	1.83E 02	1.65E 02	8.75E 01	1.62E 01	8.16E-03	0.
Total	8.77E 02	6.84E 02	5.71E 02	3.43E 02	1.88E 02	6.79E 01	5.64E 00
Kr83m	6.72E 02	4.65E 02	3.20E 02	3.37E 01	8.38E-02	1.53E-13	0.
Kr85m	1.67E 03	1.43E 03	1.23E 03	4.83E 02	4.03E 01	5.55E-04	0.
Kr85	3.67E 02	3.66E 02	3.66E 02	3.65E 02	3.63E 02	3.52E 02	2.73E 02
Kr87	3.21E 03	1.87E 03	1.08E 03	4.05E 01	6.35E-03	0.	0.
Kr88	4.56E 03	3.57E 03	2.79E 03	6.29E 02	1.19E 01	2.02E-07	0.
Kr89	4.78E 03	1.13E-02	2.14E-06	0.	0.	0.	0.
Xe131m	4.94E 01	4.92E 01	4.91E 01	4.83E 01	4.61E 01	3.77E 01	6.53E 00
Xe133m	2.44E 02	2.41E 02	2.37E 02	2.19E 02	1.77E 02	6.75E 01	1.63E-02
Xe133	8.64E 03	8.59E 03	8.54E 03	8.24E 03	7.50E 03	4.91E 03	1.26E 02
Xe135m	1.28E 03	8.87E 01	5.85E 00	4.83E-07	0.	0.	0.
Xe135	8.19E 03	7.60E 03	7.04E 03	4.46E 03	1.32E 03	5.51E 00	0.
Xe137	6.54E 03	1.54E-01	3.03E-06	0.	0.	0.	0.
Xe138	7.58E 03	4.25E 02	2.27E 01	5.23E-07	0.	0.	0.
Total	4.78E 04	2.47E 04	2.17E 04	1.45E 04	9.46E 03	5.37E 03	4.06E 02

Table 15.6-14

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE WETWELL CONTAINMENT (Ci)

ISOTOPE	1-MIN	10-MIN	1-HOUR	2-HOUR	4-HOUR	8-HOUR	12-HR	1-DAY	4-DAY	30-DAY
I-131	5.63E-01	5.63E-01	5.61E-01	5.59E-01	5.55E-01	5.46E-01	5.37E-01	5.12E-01	3.83E-01	3.14E-02
I-132	8.60E-01	8.22E-01	6.39E-01	4.73E-01	2.59E-01	7.73E-02	2.31E-02	6.19E-04	2.27E-13	1.00E-20
I-133	1.33E 00	1.32E 00	1.29E 00	1.24E 00	1.16E 00	1.02E 00	8.87E-01	5.92E-01	5.21E-02	3.74E-11
I-134	1.45E 00	1.29E 00	6.65E-01	3.01E-01	6.20E-02	2.62E-03	1.11E-04	8.35E-09	1.00E-20	1.00E-20
I-135	1.28E 00	1.26E 00	1.15E 00	1.03E 00	8.38E-01	5.50E-01	3.61E-01	1.02E-01	5.21E-05	1.00E-20
TOTAL I	5.48E 00	5.25E 00	4.30E 00	3.61E 00	2.38E 00	2.19E 00	1.81E 00	1.21E 00	4.36E-01	3.14E-02
KR-83M	6.72E 02	6.35E 02	4.63E 02	3.17E 02	1.48E 02	3.26E 01	7.15E 00	7.55E-02	1.05E-13	1.00E-20
KR-85M	1.67E 03	1.63E 03	1.43E 03	1.22E 03	8.98E 02	4.83E 02	2.60E 02	4.03E 01	5.68E-04	1.00E-20
KR-85	3.67E 02	3.67E 02	3.67E 02	3.67E 02	3.66E 02	3.66E 02	3.65E 02	3.63E 02	3.52E 02	2.70E 02
KR-87	3.21E 03	2.96E 03	1.88E 03	1.09E 03	3.66E 02	4.13E 01	4.65E 00	6.69E-03	1.00E-20	1.00E-20
KR-88	4.56E 03	4.40E 03	3.59E 03	2.81E 03	1.72E 03	6.48E 02	2.44E 02	1.30E 01	2.93E-07	1.00E-20
KR-89	4.78E 03	6.68E 02	1.19E-02	2.40E-08	1.00E-20	1.00E-20	1.00E-20	1.00E-20	1.00E-20	1.00E-20
XE131M	4.94E 01	4.94E 01	4.93E 01	4.91E 01	4.88E 01	4.83E 01	4.77E 01	4.61E 01	3.76E 01	6.38E 00
XE133M	2.44E 02	2.43E 02	2.41E 02	2.37E 02	2.31E 02	2.19E 02	2.07E 02	1.76E 02	6.60E 01	1.35E-02
XE-133	8.64E 03	8.63E 03	8.59E 03	8.54E 03	8.44E 03	8.24E 03	8.05E 03	7.49E 03	4.89E 03	1.21E 02
XE135M	1.28E 03	8.60E 02	9.39E 01	6.58E 00	3.23E-02	7.81E-07	1.88E-11	1.00E-20	1.00E-20	1.00E-20
XE-135	8.19E 03	8.10E 03	7.59E 03	7.03E 03	6.03E 03	4.44E 03	3.27E 03	1.30E 03	5.18E 00	1.00E-20
XE-137	6.54E 03	1.28E 03	1.51E-01	2.91E-06	1.08E-15	1.00E-20	1.00E-20	1.00E-20	1.00E-20	1.00E-20
XE-138	7.58E 03	4.89E 03	4.23E 02	2.24E 01	6.33E-02	5.03E-07	4.00E-12	1.00E-20	1.00E-20	1.00E-20
TOTAL NG	4.78E 04	3.47E 04	2.47E 04	2.17E 04	1.83E 04	1.45E 04	1.24E 04	9.43E 03	5.35E 03	3.98E 02

REPLACE WITH (NEW TABLE 15.6-15

Table 15.6-15

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	2.54E-10	8.98E-07	3.52E-06	5.04E-05	3.44E-04	2.24E-03	8.20E-03
I132	4.19E-10	1.22E-06	3.95E-06	2.07E-05	2.80E-05	2.81E-05	2.81E-05
I133	6.14E-10	2.13E-06	8.19E-06	1.04E-04	5.39E-04	1.46E-03	1.56E-03
I134	7.12E-10	1.53E-06	3.76E-06	7.66E-06	7.74E-06	7.74E-06	7.74E-06
I135	6.01E-10	1.99E-06	7.30E-06	7.15E-05	2.2E-04	2.63E-04	2.63E-04
Total	2.60E-09	7.76E-06	2.67E-05	2.55E-04	1.13E-03	4.00E-03	1.01E-02
Kr83m	1.99E-06	5.54E-03	1.72E-02	7.48E-02	9.03E-02	9.04E-02	9.04E-02
Kr85m	4.94E-06	1.58E-02	5.62E-02	4.64E-01	1.03E 00	1.11E 00	1.11E 00
Kr85	1.08E-07	3.84E-03	1.51E-02	2.19E-01	1.55E 00	1.15E 01	9.29E 01
Kr87	9.53E-06	2.38E-02	6.68E-02	1.99E-01	2.11E-01	2.11E-01	2.11E-01
Kr88	1.35E-05	4.07E-02	1.37E-01	8.40E-01	1.30E 00	1.32E 00	1.32E 00
Kr89	1.52E-05	7.28E-04	7.28E-04	7.28E-04	7.28E-04	7.28E-04	7.28E-04
Xe131m	1.46E-07	5.16E-04	2.03E-03	2.91E-02	2.01E-01	1.36E 00	6.01E 00
Xe133m	7.22E-07	2.53E-03	9.88E-03	1.36E-01	8.48E-01	3.95E 00	6.07E 00
Xe133	2.56E-05	9.02E-02	3.53E-01	5.01E 00	3.36E 01	2.03E 02	5.44E 02
Xe135m	3.86E-06	2.89E-03	3.70E-03	3.81E-03	3.81E-03	3.81E-03	3.81E-03
Xe135	2.42E-05	8.17E-02	3.06E-01	3.33E 00	1.21E 01	1.81E 01	1.81E 01
Xe137	2.06E-05	1.42E-03	1.42E-03	1.42E-03	1.42E-03	1.42E-03	1.42E-03
Xe138	2.28E-05	1.54E-02	1.90E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02
Total	1.44E-04	2.85E-01	9.88E-01	1.03E 01	5.10E 01	2.40E 02	6.70E 02

Table 15.6-16

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS) RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	1.2E-3 ~~2.2E-4~~	1.4E-6 ~~5.5E-6~~
*		
Low Population Zone	8.7E-4 ~~5.8E-4~~	2.9E-6 ~~6.6E-6~~
*		

Table 15.6-17

SEQUENCE OF EVENTS FOR FEEDWATER LINE BREAK OUTSIDE CONTAINMENT

Time	Event
0 sec	One feedwater line breaks.
0+ sec	Feedwater line check valves isolate the reactor from the break.
<30 sec	At low-low water reactor level RCIC would initiate, HPCS would initiate, MSLIV closure would initiate, reactor scram would initiate and recirculation pumps would trip.
~2 min	The safety/relief valves would open and close and maintain the reactor vessel pressure at approximately 1100 psig.
1-2 hr	Normal reactor cooldown procedure established.

*Applicant to Supply

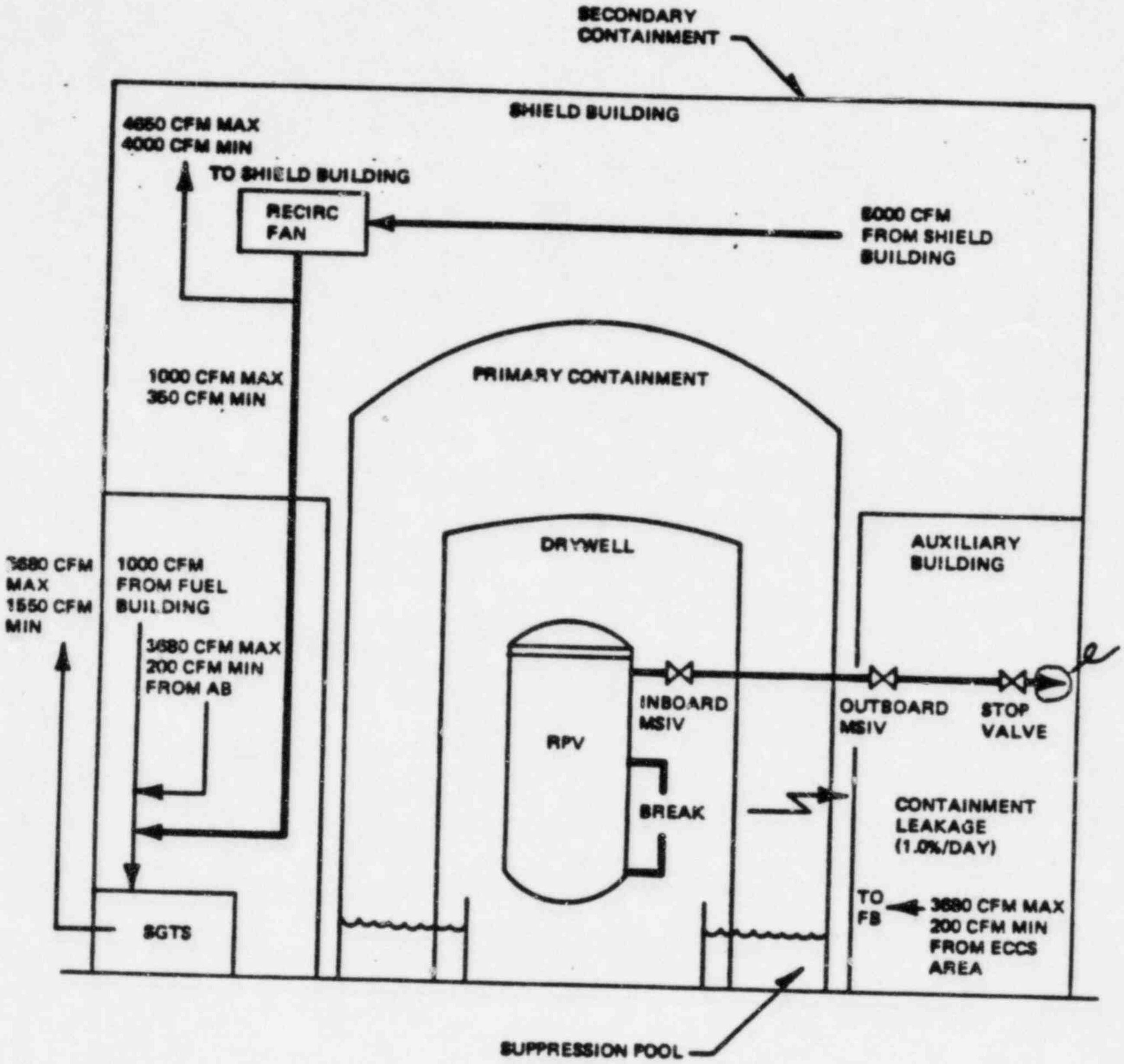

Figure 15.6-2. Post-LOCA FISSION PRODUCT RETENTION SYSTEMS ~~Leakage Pathways~~

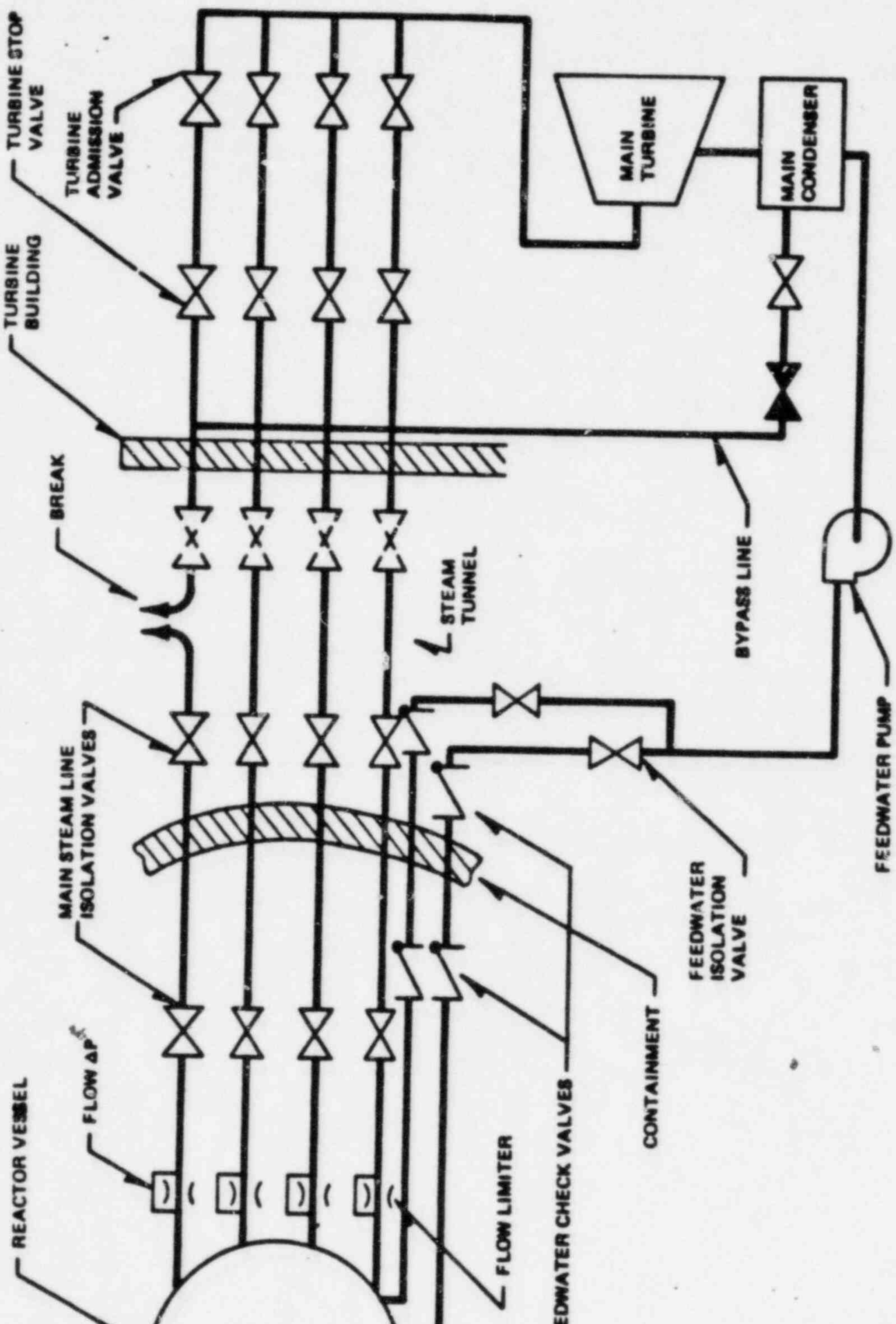

Figure 15.6-1. Steam Flow Schematic for Steam Break Outside Containment

Table 15.6-15

LOSS-OF-COOLANT ACCIDENT (REALISTIC ANALYSIS) ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

ISOTOPE	1-MIN	10-MIN	1-HOUR	2-HOUR	4-HOUR	8-HOUR	12-HR	1-DAY	4-DAY	30-DAY
I-131	5.74E-09	5.88E-08	3.98E-07	9.00E-07	1.42E-06	2.69E-06	4.13E-06	9.30E-06	6.27E-05	3.50E-04
I-132	8.79E-09	8.80E-08	5.25E-07	1.02E-06	1.34E-06	1.68E-06	1.80E-06	1.86E-06	1.86E-06	1.86E-06
I-133	1.36E-08	1.39E-07	9.26E-07	2.06E-06	3.17E-06	5.68E-06	8.21E-06	1.53E-05	3.86E-05	4.26E-05
I-134	1.49E-08	1.43E-07	7.05E-07	1.11E-06	1.25E-06	1.29E-06	1.29E-06	1.29E-06	1.29E-06	1.29E-06
I-135	1.30E-08	1.32E-07	8.57E-07	1.83E-06	2.69E-06	4.26E-06	5.46E-06	7.41E-06	8.56E-06	8.56E-06
TOTAL I	5.60E-08	5.61E-07	3.41E-06	6.92E-06	9.87E-06	1.56E-05	2.09E-05	3.52E-05	1.13E-04	4.05E-04
KR-83M	7.88E-04	7.85E-03	4.53E-02	8.47E-02	1.08E-01	1.28E-01	1.33E-01	1.34E-01	1.34E-01	1.34E-01
KR-85M	1.95E-03	1.98E-02	1.25E-01	2.61E-01	3.72E-01	5.48E-01	6.58E-01	7.84E-01	8.17E-01	8.17E-01
KR-85	4.29E-04	4.40E-03	2.98E-02	6.76E-02	1.06E-01	2.04E-01	3.15E-01	7.28E-01	5.74E 00	8.31E 01
KR-87	3.77E-03	3.71E-02	2.00E-01	3.48E-01	4.17E-01	4.55E-01	4.60E-01	4.61E-01	4.61E-01	4.61E-01
KR-88	5.35E-03	5.38E-02	3.28E-01	6.54E-01	8.89E-01	1.17E 00	1.30E 00	1.38E 00	1.39E 00	1.39E 00
KR-89	6.25E-03	2.87E-02	3.25E-02	3.25E-02	3.25E-02	3.25E-02	3.25E-02	3.25E-02	3.25E-02	3.25E-02
XE131M	5.78E-05	5.92E-04	4.01E-03	9.07E-03	1.43E-02	2.72E-02	4.18E-02	9.49E-02	6.73E-01	4.89E 00
XE133M	2.85E-04	2.92E-03	1.97E-02	4.43E-02	6.91E-02	1.29E-01	1.94E-01	4.09E-01	1.89E 00	3.48E 00
XE-133	1.01E-02	1.04E-01	7.00E-01	1.58E 00	2.48E 00	4.70E 00	7.18E 00	1.60E 01	9.93E 01	3.92E 02
XE135M	1.53E-03	1.30E-02	3.60E-02	3.93E-02	3.94E-02	3.94E-02	3.94E-02	3.94E-02	3.94E-02	3.94E-02
XE-135	9.59E-03	9.77E-02	6.40E-01	1.39E 00	2.08E 00	2.45E 00	4.62E 00	6.97E 00	9.41E 00	9.42E 00
XE-137	8.39E-03	4.31E-02	5.21E-02	5.21E-02	5.21E-02	5.21E-02	5.21E-02	5.21E-02	5.21E-02	5.21E-02
XE-138	9.09E-03	7.53E-02	1.96E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01	2.10E-01
TOTAL NG	5.76E-02	4.68E-01	2.41E 00	4.78E 00	6.87E 00	1.11E 01	1.52E 01	2.73E 01	1.20E 02	4.98E 02