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TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-356
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I. DESCRIPTION OF THE PROPOSED CHANGE

In summary, utilizing the revised accident source term for light-water nuclear power plants contained in NUREG-1465, the proposed change increases the allowable leak rate specified in TSs from 11.5 standard cubic feet per hour (scf/hr) for any one main steam isolation valve (MSIV) to 100 scf/hr for any one MSIV with a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines. If the leakage rate of 100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 sch/hr through all four main steam lines is exceeded, repairs and retest shall be performed to correct the condition.

The specific proposed changes are listed below:

A. The surveillance requirement for the primary containment is delineated in Surveillance Requirement 4.7.A.2.g (TS Page 3.7/4.7-7 for Units 1 and 3).

   The third paragraph of the current surveillance requirement states, in part:

   > "The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours."

   The proposed surveillance requirement states:

   > "The total leakage from all penetrations and isolation valves, except the main steam isolation valves, shall not exceed 60 percent of L_a per 24 hours."

B. The surveillance requirement for the primary containment is delineated in Surveillance Requirement 4.7.A.2.g (TS Page 3.7/4.7-7 for Unit 2).

The third paragraph of the current surveillance requirement states, in part:

> "The total path leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours."

The proposed surveillance requirement states:

> "The total leakage from all penetrations and isolation valves, except the main steam isolation valves, shall not exceed 60 percent of L_a per 24 hours."

C. The surveillance requirement for the Main Steam Isolation Valves is delineated in Surveillance Requirement 4.7.A.2.i (TS Page 3.7/4.7-8 for Units 1, 2 and 3).

The current Surveillance Requirement states:

> "The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 11.5 scf/hr for any one main steamline isolation valve is exceeded, repairs and retest shall be performed to correct the condition."

The proposed Surveillance Requirement states:

> "The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines is exceeded, repairs and retest shall be performed to correct the condition."

II. **REASON FOR THE PROPOSED CHANGE**

A. HISTORICAL TREATMENT OF MSIV LEAKAGE RATES

The purpose of controlling the MSIV's leakage rate is to ensure isolation of the reactor coolant system in the event of a break in a steam line outside the primary containment, a design basis loss of coolant accident (LOCA), or other events requiring containment isolation. Although the MSIVs are designed to provide a leak tight seal, operating experience has shown that these valves invariably exhibit some level of minor leakage. The current TS allowable MSIV leakage rate is extremely limiting and routinely requires repair and retest of the MSIVs even with these minor levels of leakage. This can significantly impact the maintenance work load during plant outages, contribute to outage extensions, and increase the radiation dose to maintenance workers. The outage planning group typically schedules several days of contingency time for repair and retest of the MSIVs.

Failures of MSIVs to meet the current typical TS leakage limit have been documented in response to surveys conducted by the NRC during the early 1980s and by the Boiling Water Reactor Owners' Group (BWROG) during the middle and late 1980s. As many as 50 percent of the total "as found" MSIV local leak rate test (LLRT) results were reported in the early NRC survey to exceed the leakage rate limit.

The BWROG studied the issues regarding MSIV leakage rates, their causes, and available alternatives. The results of the BWROG study are provided in NEDC-31858P, "BWROG Report for increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems," Revision 2, and are summarized in NUREG-1169. In response to Generic Issue C-8, "MSIV Leakage and LCS Failure," the BWROG has recommended corrective actions and maintenance practices to reduce MSIV leakage rates.

B. CURRENT MSIV PERFORMANCE

Despite the improvement in leakage performance, MSIV leakage rates still frequently exceed the current typical TS limit and the resultant maintenance problems, although less severe, remain as a significant issue. Furthermore, based on extensive evaluation of valve leakage data, the BWROG has found disassembling and refurbishing the MSIVs to meet very low leakage limits frequently contributes to repeating failures. In most cases, machining of the valve seat is required to meet the current TS limits. Each time the seat is machined, the thickness is reduced, leading to earlier than necessary seat replacement. Disassembly and assembly also cause wear on the various components removed and replaced. By not requiring disassembly and refurbishment of the valves for minor leakage, the utility reduces its susceptibility to one of the contributors to recurring valve leakage problems which lead to later LLRT failures and the possibility of compromising plant safety.

C. REASON FOR CHANGE IN MSIV ALLOWABLE LEAKAGE RATES

The current TS allowable MSIV leakage rate (11.5 scf/hr) is extremely small considering the valve's physical size and operating characteristics (large size and fast-acting). Additionally, the ability of the turbine building equipment to contain the radioactive material was not considered at the time the leakage limit was established. Based on the in-depth evaluation of MSIV leakages, the BWROG has concluded that leakage rates of over 500 scf/hr are not indicative of substantial mechanical defects in the valves which would challenge the capability of the valves to fulfill their safety function of isolating the steam lines. Therefore, as demonstrated in the Safety Analysis section, the proposed increased allowable MSIV leakage rate will not affect the MSIV's isolation function performance.

This proposed increase in the allowable MSIV leakage rate provides a more realistic, but still conservative, limit for the MSIVs. This increase in allowable leakage will result in significantly reduced MSIV maintenance costs, reduced dose exposure to maintenance personnel, reduced radwaste generation, reduced outage durations, extend the effective service life of the MSIVs, and results in no significant impact on the health and safety of the public.

D. REASON FOR EXEMPTING MSIV'S FROM TOTAL CONTAINMENT LEAKAGE

10 CFR 50, Appendix J, Section II.K, defines L_a as the maximum allowable leakage rate at pressure P_a as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license (Note that the BFN P_a is 49.6 psig). Section 5.2.4.5 of the BFN Updated Final Safety Analysis Report (UFSAR) states that L_a was defined as the leakage rate of 2.0 percent per day of the free volume of the primary containment. UFSAR Section 14.6.3.5 states that the primary containment free volume is 283,000 ft^3 and that two percent of this volume per day produces a leakage rate of 235 cfh at 49.6 psig (~670 scf/hr).

Increasing the allowable MSIV leakage to a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines would reduce the allowable leakage from other containment isolation valves. The two primary drywell leakage paths (into the reactor building, through SGTS and out the stack or the MSIV leakage through the condenser and out the turbine building) have been separately analyzed and the resulting doses found to be acceptable. An exception to adding the MSIV leakage to the total allowable containment leakage is necessary to support a reasonable containment isolation leakage testing and control program.

E. HISTORICAL TREATMENT OF SOURCE TERMS

For currently licensed BWRs such as BFN, the characteristics of the fission product release from the core into the containment are set forth in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." These requirements were derived from the 1962 report, TID-14844. The prescribed release consists of 100% of the core inventory of noble gases and 50% of the iodides (half of which are assumed to deposit on interior surfaces essentially instantaneously). These values were based largely on experiments performed in the late 1950s involving heated irradiated UO_2 pellets. TID-14844 also included 1% of the remaining solid fission products, but these were dropped from consideration in Regulatory Guides 1.3 and 1.4. The 1% of the solid fission products are considered in certain areas such as equipment qualification. Regulatory Guide 1.3 specifies that the source term within containment is assumed to be instantaneously available for release and that the iodine chemical form is assumed to be predominantly (91%) in elemental form, with 5% assumed to be particulate iodine and 4% assumed to be in organic form.

F. REVISED SOURCE TERM

The development of revised radiological accident source terms is an outgrowth of over 20 years or more realistic accident source term studies beginning with the Reactor Safety Study (WASH-1400) of 1975. This work has involved nuclear power industries and regulatory agencies worldwide. In the U.S. the work has included the post-TMI-2 accident studies, the IDCOR Program, studies of the Chernobyl-4 accident, and the publication of NUREG-0956 and NUREG-1150. NUREG-1150 may be viewed as the successor to WASH-1400.

Source term estimates under severe accident conditions became of great interest shortly after the TMI-2 accident when it was observed that only a relatively small amount of iodine was released to the environment compared to the amount predicted to be released in licensing calculations. The current effort to revise the Design Basis Accident (DBA) source term to replace

TID-14844, its associated implementation guidance (e.g., Regulatory Guides 1.3 and 1.4, and the source term specifications contained in the Standard Review Plan), for the 10 CFR 100 DBA as well as other DBAs started in 1990. This effort arose out of the technical initiative of the Advanced Light Water Reactor (ALWR) Program (under EPRI management) and the technical and regulatory interaction between the ALWR Program and the NRC. The ALWR Program, with the support of the U.S. Department of Energy (DOE) Advanced Reactor Severe Accident Program (ARSAP), published a preliminary report in September, 1990 for the Evolutionary ALWRs and a more extensive study for the passive Advanced Boiling Water Reactors (ABWRs) in February, 1991. The results of this second study, together with the subsequent NRC source term work formed the basis for the source term specification for passive plants in Volume 3 of the ALWR Utility Requirements Document.

The distillation of accident source term knowledge by the NRC Staff (as gathered together in NUREG-1150) led to the publication in June 1992 of Draft NUREG-1465, the initial NRC proposal for a revised DBA source term. The final-1465 was published in February, 1995.

Much progress has been made in achieving consensus between the revised DBA source terms in NUREG-1465 and the ALWR Program. In both cases the artificial distinction between the character of DBAs and of severe accidents has been eliminated. The revised DBA source term of Draft NUREG-1465 was applied by ABB-Combustion Engineering to the design certification of the System 80+ standard plant. This effort was successfully completed with a Final Design Approval being issued in August 1994 with no open items. The beneficial impacts of applying the revised DBA source term to System 80+ included a 50% increase in the allowable containment leak rate and removal of the safety-grade designation for charcoal filters in all safety-related trains except the small control room units. Work on the ALWR source term is continuing with the AP600 design certification in which the final NUREG-1465 is being applied.

In large part as a result of this experience in developing and applying the revised source term to the ALWR, there is growing interest in its application to operating plants. This is evidenced by a recent NRC letter to NEI, dated July 27, 1994, stating that some features of the new source term could lead to relaxation of certain operational requirements, while others could lead to safety enhancements. The ACRS has also encouraged use of the revised source term on operating plants in a letter to the Commission, dated September 20, 1995.

The revised source term in NUREG-1465 is expressed in terms of start time and duration of release of fission products into the containment, types and magnitudes of the species released, and other important attributes such as the chemical forms of iodine. This mechanistic approach presents a more realistic, but still conservative, characterization of the fission product source term present in the containment from a core damage accident. This characterization of the source term has impacted the design of engineered safety features for ALWRs and is also expected to impact these features for operating plants.

Work done in conjunction with development of the revised source term has shown that the fission product releases to containment can be categorized into phenomenological phases associated with the time of release, the degree of fuel melting and relocation, and reactor pressure vessel integrity. The release phases have been defined as follows:

- Coolant Activity Release
- Gap Activity Release
- Early In-Vessel Release
- Ex-Vessel Release
- Late In-Vessel Release

For purposes of the DBA source term, the ALWR applications use only the first three phases and in meetings with industry, the NRC has agreed that this same approach should apply to operating plants. This is reasonable based on past operational and accident experience, and information from modern plant designs and emergency operating procedures, which indicate that many cooling systems must fail for core damage to occur. It is likely that one or more of these systems will be returned to service before core damage progresses to the point of reactor vessel lower head failure.

NUREG-1465 also provides general guidance on fission product removal mechanisms. The BFN revised source term utilizes specific fission product removal models which are consistent with this general guidance and with the NUREG-1465 characterization of the source term released into containment.

III. SAFETY ANALYSIS

A. DESCRIPTION OF THE FACILITY

A diagram depicting the general plant layout is provided in BFN UFSAR Figure 12.2-52 (Figure 1). It shows the relative location of the Reactor and Turbine Buildings to the plant stack. An overall layout of the BFN site is provided in UFSAR Figure 2.2-4 (Figure 2).

Dimensional information is provided on this figure and in Table 1. The offsite dose is calculated assuming that a member of the general public is located at the nearest edge of the exclusion area boundary (EAB) and low population zone (LPZ). The control rooms are located on the top floor of the Control Building. The Control Building is a three story structure that is located between the Reactor and Turbine Buildings. A transverse section of the Reactor, Control, and Turbine Buildings is provided on UFSAR Figure 1.6-8, Sheet 2 (Figure 3).

The air intakes for the Control Room Emergency Ventilation System (CREVS) are located to the northwest and southeast sides of the turbine building (Figure 4) to reduce the concentration of effluents being introduced into the control bay habitability zone (CBHZ). The CBHZ is located on the top floor of the Control Building. The CBHZ contains the Units 1, 2, and 3 control rooms, equipment rooms, relay room, lunch room, rest rooms, and office spaces.

The Control Building Heating, Ventilating, and Air Conditioning (HVAC) System is depicted in BFN UFSAR Figure 10.12-2a. A simplified schematic of the system is provided as Figure 5. The Control Bay ventilation towers, located on the north wall of the reactor building, provide the outside air for the Control Building supply ductwork. Ventilation supply fans, which are located in the ventilation towers, pressurize this supply ductwork, including the ductwork that traverses the main control bay habitability zone. Some of the fans operate during the accident recovery period (30 days) to supply necessary cooling for essential equipment. In addition, the cable spreading room ventilation system, while not required to operate after an accident, is not prohibited from functioning after an accident. This could also contribute to the unfiltered inleakage into the habitability zone.

The CREVS is activated by an accident signal or high radiation signal from the Control Building intake duct radiation monitors. The same signals also initiate the isolation of the CBHZ. The CREVS processes outside air needed to provide ventilation and pressurization for the CBHZ during isolated conditions. The two 100 percent redundant filter trains are safety-related and are powered from separate divisions of normal and emergency diesel power. Only one train operates post accident with the other train on standby.

Each train of the CREVS is designed to process 3,000 cfm of outside air for 30 days without danger of saturation. The filtered outside air aids in pressurizing the CBHZ to greater than 1/8 in. W.G. with respect to the outdoors. Outside air for the CREVS is drawn from both of the main outside air intake ducts supplying ventilation tower 1 and ventilation tower 3. Outside air pulled from these two intakes passes through a High Efficiency Particulate Adsorber (HEPA) filter bank located in ventilation tower 2.

During accident conditions, this air supply to the control room and relay room air handlers is isolated. However, the fans that provide this air continue to operate in order to supply cooling to the various mechanical equipment spaces and make-up to the air handlers serving the lower floor. The outside air is provided by the Board Room Supply Fans located in the Units 1 and 3 vent towers. These fans have the following capacity:

Unit 1A Board Room Supply Fan - 14,400 CFM
Unit 1B Board Room Supply Fan - 13,400 CFM
Unit 3A Board Room Supply Fan - 10,260 CFM
Unit 3B Board Room Supply Fan - 10,260 CFM

The cable spreading room supply and exhaust ductwork traverses the habitability zone. The fans associated with the cable spreading room have the following capacities:

Units 1 and 2 Supply Fan - 5,000 CFM
Unit 3 Supply Fan - 5,000 CFM
Common Exhaust Fans - 10,000 CFM

The exhaust system that serves the electrical equipment spaces on Elevation 593 has the following fan capacities:

Units 1 and 2 Exhaust Fans - 5,700 CFM
Unit 3 Exhaust Fan - 1,700 CFM

There are two toilet exhaust fans for the control room areas. They have the following capacities:

Units 1 and 2 Exhaust Fans - 550 CFM
Unit 3 Exhaust Fan - 330 CFM

The shutdown board room exhaust fans, which are located in the Units 1 and 3 vent towers, have been abandoned in place and will only be used for smoke removal, if necessary.

B. SOURCE TERM

The accident source term for light-water nuclear power plants from NUREG-1465 was used for these analyses. NUREG-1465 describes radionuclide release from the fuel in four phases: gap, early in-vessel, ex-vessel and late in vessel. Only the first two phases are considered for design basis accident applications. The two release phases are referred to as the gap release phase and the fuel release phase, with the fuel release phase making use of only the early in-vessel contribution from NUREG-1465.

The application of the source term of NUREG-1465 requires the identification of the plant type (PWR or BWR) and a decision as to the time of the start of the gap release. The start of the gap release is chosen as 30 seconds, which is an appropriate time for the start of the gap release for a PWR. In general, the PWR gap release is expected to occur much more rapidly than the BWR gap release so the time of the start of the gap release used in this analysis is conservative. Once begun, the gap activity release is uniform for a duration of 30 minutes. Note that at the end of the fuel release phase (7230 seconds after accident initiation) the activity release is terminated due to quenching of the core (see Section 3.0)

The fraction of core inventory for each release phase is given below:

FRACTIONAL RELEASES INTO CONTAINMENT
IN TIME INTERVAL

Time Interval (sec)	Noble Gases	Iodine[1]	Cesium	Tellurium[2]	Other[3]
0 to 30	0.00	0.00	0.00	0.00	0.00
30 to 1830	0.05	0.05	0.05	0.00	0.00
1830 to 7230	0.95	0.25	0.20	0.05	0.01
7230 to End	0.00	0.00	0.00	0.00	0.00

Notes:

1. Iodine chemical composition:
   Particulate Iodine (CsI): 0.95
   Elemental Iodine (I_2): 0.0485
   Organic Iodine: 0.0015

2. Te-132 is treated as elemental I-132 except for half-life which corresponds to Te-132. This is to account for the possible removal of particulate Te-132 (by deposition and on filters) and its subsequent re-evolution as elemental I-132 upon decay. By treating the Te-132 as elemental I-132 from the beginning (with Te-132 half-life), the same amount of

I-132 activity is released as would be the case in a mechanistic model of the process described, but the release occurs much more rapidly. This means that more adverse X/Q values, breathing rates and control room occupancy factors are used in calculating the thyroid dose contribution of Te-132 than would be the case with a mechanistic model.

3. The impact of non-noble gas and non-radioiodine components of the release on the 10 CFR 100 and General Design Criterion (GDC) 19 dose calculations has been assessed in two ways: (1) the important isotopes of radiocesium and Te-132 have been included explicitly and (2) the other radionuclides have been approximated using a one percent release to the containment atmosphere (as was the case for the TID-14844 source term, the exception is that these radionuclides are subsequently treated as aerosol and released to the environment accordingly). By doing so, the impact of the other has been overstated by about a factor of ten as compared to rigorous application of NUREG-1465.

The NUREG-1465 source term is specified in terms of fractions of core inventory. A representative core burn-up is used in determining the below listed core inventory of radioisotopes.

Isotope	Activity (10^7 Ci)	Isotope	Activity (10^7 Ci)	Isotope	Activity (10^7 Ci)
Kr-83m	1.127	Xe-131m	0.105	I-131	9.378
Kr-85m	2.351	Xe-133m	0.596	I-132	13.55
Kr-85	0.136	Xe-133	18.47	I-133	18.98
Kr-87	4.481	Xe-135m	3.761	I-134	20.81
Kr-88	6.303	Xe-135	6.610	I-135	17.78
Kr-89	7.653	Xe-137	16.55	Cs-134	2.508
Kr-90	7.554	Xe-138	15.52	Cs-137	1.503
		Other	496.7	Te-132	13.33

In Regulatory Guide 1.3, the TID-14844 source term is expanded upon and clarified in the areas of:

- The timing of the release,
- Changes to the chemical and physical form, and
- The airborne release (i.e., that available for release to the environment) has been revised to

include aerosols in addition to radioiodine and noble gas.

The NUREG-1465 source term does the following in relation to the current 10 CFR 100 design basis accident source term:

- It changes the release from one involving all activity instantaneously to one involving a progressive release.

- It changes the release from one involving a high percentage of gaseous iodine (91% elemental and 4% organic in Regulatory Guides 1.3 and 1.4, for example) to one involving a high percentage of particulate (95%), with a corresponding factor of 19 decrease in the elemental percentage (to 4.85%) and a factor of 27 decrease in the organic percentage (to 0.15%).

- It changes the percentage of the core inventory released to the containment as follows (based upon the SECY-94-300 position that the NUREG-1465 ex-vessel and late in-vessel release should be omitted for 10 CFR 100 design basis accident purposes):

  - Iodine: from 50% to 30%

  - Other: from 1% for all to 25% for alkali metals (e.g., cesium), 5% for tellurium, 2% for barium and strontium, 0.25% for noble metals (e.g., ruthenium), 0.05% for the cerium group, and 0.02% for the lanthanides

The noble gas release remains at 100% for both.

C. CONTAINMENT THERMAL HYDRAULIC CONDITIONS

NUREG-1465 accounts for a timed release of radionuclides from the fuel to the containment and discusses various mechanisms for the removal of these radionuclides from the containment atmosphere. Since radionuclide removal mechanisms are dependent on containment temperatures, pressures and flows within the containment, it is necessary to describe the thermal-hydraulic behavior of the containment to the DBA.

1. Containment Conditions

   Containment temperature and pressure is based on the DBA LOCA described in Section 14.6 of the BFN UFSAR and shown in Figure 6. After DBA initiation, containment pressure initially increases to 49.6 psig, rapidly decreases and then stabilizes at 27 psig within the first 100 seconds. Following initiation of containment cooling, containment pressure decreases to 10 psig or less for the remainder of the DBA. Since the environment in containment is saturated, containment temperature follows the same trend as containment pressure. Containment temperature initially increases to roughly 300 F rapidly decreases and then stabilizes at 275 F within the first 100 seconds. Following initiation of containment cooling, containment temperature decreases to 175 F. The peak suppression chamber water temperature remains below 173 F.

The containment temperatures and pressures given above for a LOCA DBA are correlated to the radionuclide release phases of the analysis as follows:

a. From break initiation until restoration of core cooling after the end of the fuel release phase, containment pressure and temperature are 27 psig and 275 F, respectively.

b. From the end of the restoration of core cooling until the end of the analysis, containment pressure and temperature are 10 psig and 175 F, respectively.

Since the fuel release phase is terminated by restoration of core cooling, it is consistent to maintain the containment at the elevated temperature and pressure until the end of the fuel release phase.

2. Determination of the Sweep-Out Rate

The flow from the drywell to the torus is referred to as the sweep-out rate. The sweep-out rates are used in subsequent calculations to determine radionuclide removal and the lower the estimate of the sweep-out rate, the lower the radionuclide removal. Therefore, it is conservative to underestimate the sweep-out rate. The volumetric exchange rates between the drywell and the torus during four periods were determined:

a. prior to and during the gap release phase (from 30 seconds to 1830 seconds),

b. during the fuel release phase (from 1830 seconds to 7230 seconds),

c. during the restoration of core cooling after the fuel release phase (7230 seconds to 7890 seconds), and

d. for the long-term following restoration of core cooling (7890 seconds until 30 days which is the end of the dose calculation interval).

To specify the sweep-out rate, an estimate must be made of the thermodynamic state in the drywell and the rate at which steam is produced from the core debris in-vessel, up to and including the point in time where the core-debris quench is complete.

a. Determine the Steam Production Rate During the Gap Release Phase

The gap release phase of the accident occurs from 30 seconds until 1830 seconds after the start of the DBA. Following initiation of the DBA (recirculation suction large break LOCA) the water remaining in the vessel is conservatively estimated to be the volume below the bottom of active fuel, depressurized at constant enthalpy to atmospheric pressure with steam being released from the vessel. This approach yields a conservatively small value for the water mass remaining in the bottom of the vessel after blowdown. All water above the bottom of the core is removed at its operating state with no change in phase and no liquid remaining. Then, the remaining coolant is flashed all the way down to atmospheric pressure (In reality, coolant would flash throughout the vessel as the vessel depressurizes, leaving more liquid in the bottom of the vessel than calculated, enough to at least partially cover the core). Therefore, during the gap release phase:

(1) the reactor vessel has blown down completely,

(2) the core is completely uncovered and

(3) the fuel is heating up adiabatically

As a result, there is assumed to be no steam production during the gap release phase.

b. Determine the Steam Production Rate During the Fuel Release Phase

During the fuel release phase, the steaming rate from core debris is calculated by uniformly increasing the fraction of the core participating in the boil-off of the water mass remaining in the bottom of the vessel from zero at 1830 seconds (end of the gap release phase) to 50% at 7230 seconds (end of the fuel release phase). The debris remaining in the core region is neglected in the calculation of the steaming rate during core degradation; only the assumed 50% of the core debris which progressively relocates to the lower part of the vessel and its interaction with the residual water is included in the quantification of the steam production during the fuel release phase.

The rate at which the residual water in the bottom of the vessel would be boiled off, in reality, would be determined by several heat inputs including:

(a) Sensible heat stored in the lower reactor vessel structures and in the core debris interacting with water in the lower reactor vessel,

(b) Metal-water reaction heat from core debris relocated to the lower reactor vessel,

(c) Beta radiation from core debris relocated to the lower reactor vessel, and

(d) Gamma radiation from core debris relocated to the lower reactor vessel

The only one of the heat sources considered in the analysis is the gamma radiation from core debris relocated to the lower reactor vessel. All other heat sources are conservatively neglected. It should be noted that very little of the stored energy in vessel structures would be released during the blowdown, leaving that energy available for steam production during core degradation.

The boil-off, from 1830 seconds to 7230 seconds is divided into nine 600-second time intervals. The fraction of the core debris in the lower reactor vessel increases from 5% for Interval 1 to 45% for Interval 9. At the end of Interval 9 the total fraction relocated to the lower reactor vessel reaches 50%. The average fraction of core debris in the lower reactor vessel over the period 1830-7230 seconds is, therefore, 25%.

The gamma source strength used in the heating calculations omits 100% of the noble gas and 30% of the iodides which are released to the containment atmosphere during the gap and fuel release phases. Because the shortest half-life of the relevant isotopes is 52.5 minutes and the intervals are 10 minutes in length, decay during the intervals is neglected. However, decay prior to the start of the interval has been included.

The steam generation rate corresponding to the heat transfer to the overlaying water is determined from the h_{fg} corresponding to containment conditions of near-saturation at 27 psig. Using this value, the steam generation rates by interval are:

Interval:	1830s	2430s	3030s	3630s	4230s	4830s	5430s	6030s	6630s
10^{16} Mev/sec-Mw:	0.19	0.37	0.53	0.68	0.82	0.94	1.05	1.15	1.24
BTU/sec:	999	1945	2786	3585	4310	4941	5519	6045	6518
Lbm/sec steam (Based on 932 BTU/lbm)	1.1	2.1	3.0	3.8	4.6	5.3	5.9	6.5	7.0

The average steam generation rate during the fuel release phase is 4.4 lbm/sec.

c. Determination of Steam Production During Restoration of Core Cooling

At the end of fuel release all of the core debris is quenched, both that which has relocated to the lower part of the vessel and that remaining in the core region. The final core debris quench requires the time it takes minimum Emergency Core Cooling System (ECCS) (one core spray pump) to refill the core region, and it involves only the energy stored in the one-half of the core debris not relocated to the lower part of the vessel. Leaving one-half the core uncovered for a period of 7230 seconds (less the blowdown/core uncovery time) results in core debris left in the core region with significant stored energy. The restoration of minimum ECCS will remove this stored energy at a rate determined by the coolant injection rate (drawn from the suppression pool) and the rising water level (reflood rate). To determine the reflood rate, the ECCS injection rate must be reduced by the rate of steam production. The rate of steam production in this analysis corresponds to a low estimate of stored energy in only one-half of the core debris.

To calculate the minimum stored energy in the one-half of the core debris left in the core region, the same approach is used as was used to calculate the power produced by the core debris in the lower vessel (See Table 2). Only the gamma decay stored energy is considered. The one-half core is approximated to be adiabatic for 5400 seconds from the end of the gap release phase (1830 seconds) to the end of the in-vessel release phase (7230 seconds). Noble gases and 30% of the iodides are neglected completely.

Water injected from the suppression pool post-blowdown would not be less than 150 F. To convert some of this water to steam, all of this injection water must first be raised to saturation temperature at 275 F in the drywell. The additional water injected (that which produces steam instead of reflood inventory) is raised to saturation and then boiled off. Also, during the reflood decay power continues to be generated in the lower vessel, producing steam at a rate of 7.0 lbm/sec (see last interval steam production above), and an equivalent amount of steam will be produced due to decay power in the core region. This yields a reflood time of 660 seconds and a total steam production rate of 31.9 lbm/sec.

d. Long-term Steam Production

Once the core debris is quenched in-vessel, the production of steam and hydrogen ceases. Steam condensation in the drywell (in particular, if drywell sprays are actuated) causes a return of non-condensibles and radionuclides from the torus airspace to the drywell.

As noted below, the volumetric flows are determined by noting that, prior to the restoration of core cooling, the drywell is steam-filled at 41.7 psia and near-saturation.

- Prior to and during the gap release phase, the flow rate from the drywell to the torus is:

  0 - 1830 seconds = 0 lbm/sec = 0 cf/hr

- During the fuel release phase, the flow rate from the drywell to the torus is:

  1830 - 7230 seconds = 4.4 lbm/sec = 45 cf/sec = 1.6E+5 cf/hr

- During the core quench phase, the flow rate from the drywell to the torus is:

  7230 - 7890 seconds = 31.9 lbm/sec = 325 cf/sec = 1.2E+6 cf/hr.

- As previously mentioned, once the core debris is quenched in-vessel, the production of steam and hydrogen ceases. Steam condensation in the drywell (in particular, if drywell sprays are actuated) causes a return of non-condensibles and radionuclides from the torus airspace to the drywell. During this time period the drywell and torus airspace are considered a single well mixed volume.

The exchange rate between the drywell and the torus is assumed to be constant during the fuel release phase (from 1830 seconds to 7230 seconds). This is slightly non-conservative because it overestimates the removal rate from the drywell early in the release phase. However, it does simplify the analysis; and for relatively low removal rates (of the order of one containment volume per hour) the underestimate of the late removal compensates nearly completely for the overestimate of the early removal.

For a drywell volume of 159,000 ft^3 the quench flowrate of 1.2E+6 cf/hr corresponds to a drywell sweep out rate of 7.4 per hour. This rate is sufficiently high to permit it to be used to characterize the well-mixed behavior of the containment beyond the core debris quench.

The effect of condensation in the drywell on the correspondence between the minimum sweep out rate and the minimum steaming rates was considered (i.e., the effect of condensation would decrease the sweep-out rate for a given steaming rate). First, condensation would not be expected during core degradation because of heat-sink saturation during and immediately after blowdown. Drywell condensation could decrease the sweep-out rate; but condensation also brings about diffusiophoretic removal of aerosol. Since the NUREG-1465 source term is dominated by aerosol, this is an important effect. The diffusiophoretic aerosol removal (recognizing the drywell is steam-filled) rate is equal to the steam condensation rate divided by the steam density. This removal rate is the same as one would obtain for the volumetric sweep-out rate of the drywell if the steam generated in the drywell were flowing into the torus instead of condensing in the drywell. Therefore, the two phenomena are essentially equivalent; the radionuclide removal efficiency would be expected to be greater for diffusiophoretic deposition than for flow to the torus because of pool bypass and the difficulty of scrubbing small aerosols. Therefore, steam condensation in the drywell, to the small extent it may occur, can be neglected.

D. RADIONUCLIDE REMOVAL MECHANISMS

During the DBA, fission product aerosols are released from the damaged core into the drywell, together with significant amounts of steam and non-condensible gases. The steam and gases, as well as the heat transfer to the gases in the drywell, will cause an increase in drywell pressure and result in a significant sweeping flow into the wetwell through the vent/downcomers that connect the drywell and wetwell. Leakage flows into the main steam lines through the MSIVs and directly to the reactor building are also expected. All these flows will dilute or remove the aerosols from the drywell and, at the same time, the aerosols will experience other removal processes inside the drywell, such as sedimentation, diffusiophoresis and thermophoresis. The analysis models radionuclide removal from the containment atmosphere by various natural removal mechanisms. These removal mechanisms are listed below:

1. Removal of aerosols and elemental iodine by sedimentation in the containment

2. Removal of particulate and elemental iodine in the suppression pool due to scrubbing

3. Removal of aerosols in the main steam lines by sedimentation

4. Removal of elemental iodine in the main steam lines by sedimentation

These removal mechanisms are discussed in the following Subsections.

1. Removal of Aerosols and Elemental Iodine by Sedimentation in the Containment

   In the analysis of the removal of aerosols and elemental iodine in containment:

   a. The drywell is treated as a single calculational volume, implying that the drywell is well mixed during the DBA. This is a reasonable approximation given that non-condensible gases, fission product gases and aerosols are blowing into the drywell atmosphere, while significant heat and mass transfers are taking place.

   b. Since the drywell walls are insulated and the initial blowdown from the vessel occurs prior to the initial radionuclide release, condensation and sensible heat transfer onto the drywell walls are conservatively neglected in terms of removal of aerosols by diffusiophoresis and thermophoresis.

c. The hygroscopicity of the aerosols is ignored. This is conservative since hygroscopicity results in growth of soluble aerosols (e.g. CsI and CsOH) and enhances their removal by sedimentation.

d. The amount of non-fission product aerosols released to containment is equal to the amount of fission product aerosols. Note that a larger amount of non-fission product aerosols released to the containment would enhance overall aerosol agglomeration and, therefore, would increase aerosol sedimentation.

e. Aerosol size distribution is log normal with a geometric mean radius of 0.22 micron and geometric standard deviation of 1.81.

f. The aerosol release rates are based on the source term and are given below:

	Gap Release (g/sec) [30-1830 seconds]	Fuel Release (g/sec) [1830-7230 seconds]
CsOH	6.68898	8.74614
CsI	0.89682	1.49471
Te	0	0.32315
BaO	0	0.43419
SrO	0	0.27463
CeO2	0	0.11283
La2O3	0	0.03636
Ru	0	0.27037
Non-Fission	0	14.221

g. A conservatively small sedimentation area is used for the drywell. Moreover, the sedimentation removal rates calculated for the drywell are applied to the torus airspace after the debris quench. No sedimentation is credited in the torus airspace prior to debris quench. Due to the effects of pool bypass the mass airborne in the torus airspace at the end of quench is greater than that airborne in the drywell and the torus airspace has a smaller volume and a greater sedimentation area. Therefore, to apply drywell removal rates in the torus after debris quench is a significant conservatism.

Based on the above, the sedimentation removal rate for aerosols in the drywell (DW) and torus/wetwell (WW) are given below:

TIME (SEC)	LAMBDA DW (PER HOUR)	LAMBDA WW (PER HOUR)
0.00	0.00	0.00
30	0.35	-
2,400	0.45	-
3,200	0.55	-
4,000	0.65	-
4,885	0.75	-
6,300	0.85	-
7,360	0.95	-
7,890	-	0.95
8,570	0.85	0.85
9,840	0.75	0.75
11,760	0.65	0.65
14,530	0.55	0.55
18,650	0.45	0.45
24,980	0.35	0.35
35,570	0.25	0.25
57,220	0.162	0.162
100,000	0.00	0.00
3,000,000	0.00	0.00

h. Elemental Iodine Plateout as an Aerosol in the Drywell

This discussion was incorporated into Section 4, below.

2. Removal of Particulate and Elemental Iodine in the Suppression Pool.

The maximum removal efficiency that can be credited for the passage of particulates and of elemental iodine (including Te-132 which is being treated as elemental I-132 except for half-life) through (and around) the suppression pool is determined using the following method. In the analysis, a Decontamination Factor (DF) of 100 is used for particulate and elemental iodine which passes through the pool. To account for pool bypass, a steam mass flow corresponding to ten times the drywell-to-torus vacuum breaker surveillance test acceptance value is used (Note that a review of the surveillance test data indicates that the actual measured value is, on average, substantially below the test acceptance value). It is compared to the mass flow out of the drywell during the fuel release and during core quench to determine the bypass fraction. No credit for removal is taken for the fraction of the drywell sweep out flow which bypasses the suppression pool. The overall pool DF (expressed as a filter efficiency) is calculated accordingly to be:

TIME INTERVAL(S)	NG	PART I, Cs, OTHER	ELEM. I2	ORGANIC	TE
0 TO 7230	0	0.72	0.72	0	0.72
7230 TO 7890	0	0.95	0.95	0	0.95
7890 TO END	0	0	0	0	0

3. Aerosol Removal in the Main Steam Lines

During the DBA, aerosols suspended in the drywell may be entrained in the flow that enters the main steam lines through the MSIV leakage and experience removal processes, such as sedimentation, diffusion, diffusiophoresis and thermophoresis. Since the leakage flow is small but the size of the main steam lines is large, the bulk flow velocity (driven by the leakage flow) in the main steam line is very small. As the average velocity of the aerosols entrained in the leakage flow is the same as the bulk velocity of the flow, the average residence time for particles (i.e., the time the aerosols spend within the volume of the main steam lines) can be very long for a typical length of main steam line.

To calculate the retention of aerosols in the main steam lines, the average residence time for the aerosols is determined first. Then, the removal rates of the aerosols are calculated. Finally, integration of the removal rate over the average residence time yields the amount of aerosols removed from the total aerosols entering the main steam lines.

The determination of the aerosol removal efficiency assumed:

a. The normal operating temperature of the main steam lines remains constant. Ignoring the temperature drop in the steam lines results in a smaller decontamination factor.

b. The gas flow in the main steam lines which carries the aerosols is plug flow (i.e., uniform flow along the length of the main steam line with the velocity based on the volumetric flow from MSIV leakage).

c. Aerosol sedimentation is the only removal mechanism for aerosols in the main steam lines. Deposition in the drain line and in the main condenser is conservatively ignored.

d. Aerosol size distribution is assumed to be lognormal, with a geometric mean radius of 0.22 micron and a geometric standard deviation of 1.81.

e. The steam line flowrate used to assess aerosol deposition in the steam line is based on 100 scf/hr per line (total flow 400 scf/hr). However, the overall limit on MSIV leakage is 250 scf/hr so aerosol deposition is underestimated.

f. The aerosol deposition between the inboard and outboard MSIVs was conservatively ignored.

The resulting aerosol removal efficiency for the main steam lines is determined to be 0.9985.

4. Removal of Elemental Iodine in the Main Steam Lines

Elemental iodine (i.e., I_2), released from the damaged core, plates out on the aerosol suspended in the drywell atmosphere and is transported with the aerosol. Thus I_2 leaks with the aerosol through the MSIVs and deposits on the steam line pipe wall (with the aerosol). A fraction of this I_2 is assumed to resuspend as organic iodide and is then available for transport to the environment. The fraction of I_2 that resuspends as organic is estimated where the resuspension fraction is converted to an effective filter efficiency for I_2 entering the steam lines.

To determine the effective filter efficiency the analysis first evaluates the plate-out of I_2 on aerosol; then, compares the resuspension rate of I_2 with the fixation rate in order to determine the fraction of deposited I_2 which resuspends over time; and finally, converts the resuspended fraction to a filter efficiency.

Since the surface area of the aerosol was conservatively estimated to be six times the available surface area of the drywell, it is reasonable to conclude that the I_2 will tend to plate out almost entirely on the aerosol. A second consideration with regard to I_2 plate out on aerosol is that the aerosol gradually is removed from the drywell and thus its effective plate out area decreases with time. However, the I_2 plate out rate constant (~1.7/hr) is larger than the sedimentation rate constant of the aerosol (0.3 to 0.9/hr). While the aerosol sweep-out rate constant is somewhat larger, sweep out will remove both aerosol and I_2. Thus the I_2 will plate-out on the aerosol much faster than the aerosol itself is removed from the drywell.

On the basis of the large aerosol surface area and because the I_2 will plate-out on the aerosol much faster than the aerosol itself is removed, it is reasonable to approximate that all the I_2 deposits on the aerosol and thus that the I_2 behaves as an aerosol up to the point it deposits in the steam lines.

Essentially all the aerosol which leaks through the MSIVs and into the steam lines deposits on the pipe walls. Thus the I_2 attached to this aerosol is also deposited on the pipe walls and some fraction of this I_2 resuspends. This fraction is estimated by comparing the rate constant for fixation with the rate constant for resuspension and works out to be about ½ of the deposited I_2 resuspends. Therefore, the effective filter efficiency on I_2 entering the steam lines is 0.5. Note that treating the resuspension as a filtering process is conservative because the actual resuspension occurs over a several day period, whereas the filtering process assumes that the release is instantaneous at the time of deposition on the steam lines. This filter efficiency of 0.5 is also conservatively applied to the tellurium.

E. SUPPRESSION POOL pH CONTROL TO PREVENT IODINE RE-EVOLUTION

Inorganic iodine in the containment atmosphere is removed by the previously described various natural mechanisms. To assure that the inorganic iodine remains in the suppression pool water (and the water on the floor of the drywell) and does not re-evolve as organic iodide, the pH of the water in containment must be controlled. A pH of six was assumed in the dose calculation. TVA will maintain the pH above 7.0 to provide added conservatism and operational margin. The initial pH of the suppression pool is 6.0 and during the course of the accident the pH of the suppression pool can decrease due to the radiolysis of cable (PVC and Hypalon). The method of pH control has not been finalized and will be addressed in a subsequent submittal.

1. Containment Water pH to Prevent Iodine Re-evolution

The water volume which could ultimately dissolve the iodine released from the core is the suppression pool volume and the volume of reactor vessel, recirculation loops, and Low Pressure Coolant Injection (LPCI) loops or 4.17E6 liters. For a high burn-up core, the core iodine mass is approximately 7.5 grams per Mw. The core power is 3458 Mw(t). This means the iodine mass is approximately 2.6E4 grams. The iodine core inventory (most of which is stable or near-stable iodine) would be approximately 6.2E-3 grams per liter if 100 percent were released. The NUREG-1465 source term, however, involves only a 30% release of iodine for a BWR; and therefore, the iodine concentration (taken to be I^-) is 1.87E-3 grams per liter or about 1.4E-5 gm-atoms per liter.

If $H^+ = 10^{-6.0}$ (i.e., pH = 6.0), then:

$I_2 = (H^+)^2(I^-)^2/[d + e(H^+)]$ where:

d = 4.22E-14, and
e = 1.47E-9

I_2 in the liquid phase = 4.5E-9 gm-moles/liter

I in the liquid phase = 9.0E-9 gm-atoms/liter

Since I^- in the liquid phase = 1.45E-5 gram-atoms/liter, then I/I^- = 6.2E-4 in the liquid phase.

The partition coefficient is:

$\log_{10}$ PC(I) = 6.29 - 0.0149T, where T is in K

Using the maximum pool temperature, 173 F = 352 K

PC(minimum) = 11.1 (i.e., the minimum concentration of iodine, as I_2, in the liquid phase is 11.1 times that in the gas phase. A lower temperature would yield a higher PC)

Since the gas phase volume is equal to the volume of drywell plus the volume of torus airspace (283,000 ft^3), and because the volume of the liquid phase is the suppression pool volume (127,800 ft^3), the ratio of the gas phase volume to the liquid phase volume is 2.2:1. This means that once removed from the gas phase, the mass of iodine, as I_2, in the liquid phase would never be less than that in the gas phase (11.1/2.2 = 5). Since the maximum mass ratio of I/I^- in the liquid phase is 6.2E-4, the maximum mass ratio of I in the gas phase to I^- in the liquid phase is 6.2E-4/5 or 1.2E-4. This means that the minimum ultimate DF of iodine for this system is approximately 1/1.2E-4 or 8,000, if the iodine can be removed from the gas phase initially.

As previously discussed, 0.0015 of the iodine released to containment must be considered to be organic. This fraction is 13 times larger than the fraction of the iodine released which could re-evolve as I_2 (as calculated above). Therefore, as a practical matter, there is no need to limit the removal of inorganic iodine in the analysis; the organic iodine (which is not removed by deposition or pool scrubbing) will always dominate. The water in the drywell and that in the suppression pool will have the same pH and radioiodine concentration; therefore, the concentration ratio (I_2 in the gas phase to I^- in the liquid phase) will be the same. This means that the I_2 concentration in the gas phase of the torus and the drywell will be the same, and a single control volume model of the containment is acceptable in the long-term from the standpoint of the potential for iodine re-evolution.

The minimum justifiable long-term DF for iodine in both the drywell and the torus is 8,000. If this degree of decontamination can be achieved by removal mechanisms, then the associated re-evolved I_2 will not exceed eight percent of the organic iodine in the source term specification.

2. Fraction of Aerosols Deposited in Suppression Pool Water

To determine the amount of radiolysis (of water and cable material) occurring in the containment, it is necessary to determine the fraction of radionuclide aerosols which are deposited in the suppression pool water. The analysis is based on the behavior of the aerosols in the drywell and assumes a well mixed drywell with uniform sedimentation flux and a water covered drywell floor. The fraction of the total released aerosols that suspends or has deposited on non-floor surfaces (e.g., the surface above the water covered drywell floor) is given by the following expression:

F = (Suspended Aerosol Mass + Diffused Aerosol Mass + γ*Settled Aerosol Mass)/Total Released Aerosol Mass

Where,

Diffused aerosols are those deposited on walls in the drywell,

Settled aerosols are those on all projected horizontal surfaces,

γ is the ratio of the floor area to the total sedimentation area of the drywell, and

The fraction of aerosols which eventually end up in the water is given by: $P = 1 - F$. In this fraction, the leaked aerosols are included. However, because the amount of the leaked aerosols is much less than the amount of swept aerosols, P is a reasonable approximation.

Based on the above, the average fraction of the released aerosols which are in the suppression pool water is 0.79.

3. Determination of Suppression Pool pH During the DBA.

The analysis of the pH of the suppression pool during the DBA:

a. Calculates the [OH-] or [H+] concentration in the suppression pool due to radiolysis of water. In the analysis, 90% of the radionuclide Cs is in the form of CsOH and the balance is CsI. Likewise, 5% of the radionuclide I is in the form HI and the balance is in the form CsI. Over time, the [OH-] from CsOH is reduced by the production of HNO_3 from the radiolysis of water.

b. Calculates the [HCL] in the suppression pool as a result of the radiolysis of electrical cable for both PVC and Hypalon.

c. Determines the [H+] added to the suppression pool from 1 and 2 above.

d. Determines the pH of the suppression pool.

It was determined that suppression pool pH exceeds 7.0 for at least the first two hours of the accident without any pH control measures(as previously discussed, the method of long-term pH control has not been finalized and will be addressed in a separate submittal).

F. MODELING OF RELEASE PATHS

1. Releases from Primary Containment

Section 5.2.4.5 of the BFN UFSAR states that L_a was defined as the leakage rate of 2.0 percent per day of the free volume of the primary containment. This is also the maximum allowable leakage rate specified by Technical Specification 3.7.A.2.b. This leakage is composed of:

- Drywell Leakage: Equal to 2% of the drywell volume per day (132.5 cf/hr)- leakage is to the reactor building. This is the maximum allowable leakage rate specified by Technical Specification 3.7.A.2.b.

- Torus Airspace Leakage: Equal to 2% of the torus airspace volume per day (103.3 cf/hr)- leakage is to the reactor building

- Containment Atmosphere Dilution (CAD) operation: Equal to 8340 cf/hr from the torus airspace intermittently over 30 days (containment is well mixed by the time of CAD operation so a drywell source would be equivalent) - all of the flow is through the Standby Gas Treatment System (SGTS) filters, 99.97% to the stack, 0.3% to the stack room (i.e., stack bypass). The CAD flow rates are:

START TIME (DAYS)	FLOW (CF/HR)
0.00	0.00
10	8,340
11	0.00
20	8,340
21	0.00
29	8,340
30	0.00

- Torus hardened Wetwell Vent Leakage: Equal to 10 cf/hr from the torus airspace - leakage is to the stack.

The drywell, torus airspace and CAD releases are processed by the SGTS. The primary containment releases are routed to the stack for release. The flow rate is equal to 1.32E6 cf/hr. The SGTS flow is from the reactor building, through the SGTS filters, to the top of stack (except for 300 cf/hr which goes to the stack room as bypass and is modeled as a ground level release). This is for three SGTS trains in operation, which was determined to be the limiting case.

There were two release points modeled for the plant stack (chimney), the top and the bottom. Details of the modeling of each release path is provided below:

a. Top of Stack

The reinforced concrete Seismic Class I plant stack is shown in UFSAR Figure 12.2-52. An excerpt is provided as Figure 7. The stack is 600 feet tall. The base of the stack is located at plant Elevation 568. The top of the stack is located at plant Elevation 1168. The CREVS air intakes are located at 83° and 99° clockwise from true north from the plant stack, respectively. The distance between the intakes and the plant stack and the dimensions of the top of the stack are included in Table 1. A flow rate from all three trains of the SGTS of 22,000 cfm was assumed. The release temperature was assumed to be at ambient.

b. Base of Stack

The dampers used to isolate potential ground level release paths from the stack are safety related. They automatically close to prevent a backdraft through the ductwork. Leakage from the stack room was assumed to be equal to 300 cf/hr. The base of the stack is located at plant Elevation 568. The inside diameter of the base of the stack is 61 feet and the height of the room at the base of the stack is 31½ feet.

2. MSIV Leakage Path Through the Main Condenser

There are four main steam lines from the reactor to the MSIVs. In addition to the leakage from the primary containment to the secondary containment, the MSIVs were assumed to leak at a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines.

The primary containment leakage was assumed to flow through the MSIVs to the main condenser via the drain line pathway. The MSIV leakage release is entirely through the main condenser; there is no release considered via the high pressure turbine. The flow split between the drain line/main condenser flowpath and the high pressure turbine flowpath is about 200:1. Since deposition in the steam lines is the only deposition considered for MSIV leakage, the only mechanism that could create a difference in the calculated relative dose between the two pathways is delay in the main condenser. This effect is estimated to be of the order of a factor of 2 or 3 in dose reduction; therefore, the importance of including the high pressure turbine release would be to increase the dose by 1-1.5% (negligible).

After its release into the Turbine Building, the MSIV leakage flows through the non-safety related ventilators mounted on the Turbine Building roof. The free volume of the Turbine Building associated with each unit was 2,100,000 cubic feet. However, turbine building hold-up is conservatively neglected. The flow rate from the Turbine Building vents was 8,640,000 cfh.

There are a total of 9 Turbine Building exhaust fans per unit (27 total fans). Each fan is rated at 16,000 CFM. The fans are located at plant Elevation 682. The fans have an opening diameter of 50 in. The fans are located atop 8 in. curbs and are 32 in. high. The actual air flow release point is approximately 20 in. above the roof. The fans are non-safety related and are assumed to run continuously, since this assumption is the most conservative with respect to calculated doses. They do not have an accident operating mode.

The modeling of the MSIV leakage path used in this analysis is consistent with the intent of the Boiling Water Reactor Owner's Group (BWROG) method for processing MSIV leakage following a DBA coincident with a seismic event (NEDC-31858P, Revision 2, BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems). In accordance with the BWROG analysis, use of the main steam lines and the condenser for radionuclide attenuation requires:

a. The main steam line downstream of the outboard MSIV is used to convey MSIV leakage to the condenser.

b. The internal cross sectional area of the drain line should be nominally 1.0 sq. in. or larger.

c. The capability of establishing a main steam drain path even if off-site power is not available.

d. Verification of seismic adequacy of the main steam piping, main steam drain line and condenser.

The intent of the above requirements will be met and will be addressed in a separate submittal for each unit. TVA currently anticipates that the application of this methodology will include the need for two manual operator actions from the turbire deck within 30 to 60 minutes to ensure an analyzed flow path.

3. Atmospheric Dispersion Coefficients (X/Q)

Meteorological data for BFN, from the five year period from January 1, 1987 to December 31, 1991, was used for the calculation of the annual average sector X/Q values. The parameters used were the wind speed at the 90 m elevation, the wind direction, and the difference in temperature between the 10 and 90 m elevations (for stability classification purposes).

G. RADIONUCLIDE TRANSPORTATION AND DOSE CALCULATION

The radionuclide transport and dose analysis uses a model of BFN and its environment and accounts for airborne activity release paths from the primary containment to the environment (see Figures 8 through 11). The transport and dose analysis accounts for removal of radionuclides by natural mechanisms and active mechanisms (e.g., SGTS operation).

The model used in the radionuclide transport and dose analysis consists of 7 nodes, with eight identified regions (See Figure 8). These regions are:

- Region 1 is the environment and is not used in the activity calculations.
- Region 2 is the drywell. All source releases from the fuel are directed into Region 2. Radionuclide removal in this region is allowed via natural removal mechanisms. This is accomplished by providing a radionuclide removal time constant as a function of time.
- Region 3 is the wetwell (torus). When the vacuum breakers on the vent pipes are closed, the gas mixture driven from the drywell into the wetwell will exit at a submerged elevation within the suppression pool. Fission products traversing this path will be scrubbed (via a filter in the model) prior to entering the wetwell air space. When the vent pipe vacuum breakers are open, the gas space of the wetwell and drywell communicate directly, without further scrubbing. Radionuclide removal in the wetwell air space is considered via a table of radionuclide removal time constants.
- Region 4 is the control room. The control room receives air intake from the environment. Both filtered and unfiltered air enters the control

room. The activity entering the control room originates from:

- Reactor Building and wetwell releases through the stack,
- Stack room releases,
- Drywell releases (which are diluted in the Reactor Building, but bypass the stack), and
- Main condenser releases due to leakages through the main steam isolation valve.

- Region 5 is the Reactor Building. The Reactor Building accepts containment leakage from the drywell and wetwell air space.
- Region 6 is room at the base of the stack. Leakage may enter this room via filtered leakage originating and the Reactor Building.
- Regions 7 and 8 are only weakly coupled to the remainder of the model. Flow leaving the Drywell (Region 2) through the MSIVs enters the main steam line volume (Region 7). The activity of this flow is decreased by the decontamination factor associated with natural aerosol deposition processes in the pipe.
- Region 8 is the main condenser. Fission products enter the main condenser at a low rate and are diluted by the large air volume of the condenser.

The following radionuclide removal mechanisms are accounted for:

- Deposition in the drywell and, after core debris quench, in the torus space.
- Suppression pool scrubbing.
- Deposition in the main steam lines up to the drain connection (effective filter efficiency: particulates 0.9985; elemental iodine 0.50; organic iodine 0.0).
- Active filtration by the SGTS and CREVS, but without credit for charcoal absorbers (0.99 efficiency for HEPA removal of particulates).
- Holdup and dilution in the main condenser.

1. Summary of Basic Equations

The basic equations to be solved for the analysis are presented below. The equations apply individually for all isotopes/form combinations considered in the analysis.

The general form of the activity distribution equation is :

$$\frac{D(A(N,i,k,t))}{Dt} = S(N,i,k,t) - \lambda_d(i)A(N,i,k,t) - \lambda_s(N,k)A(N,i,k,t) - \Sigma L_{NM}A(N,i,k,t) + \Sigma L_{MN}A(M,i,k,t)F_{MN}(k)$$

Where:

A(N,i,k,t) is the activity in Region N of isotope I with physical form k. The only isotope to have multiple physical forms is iodine that has three: elemental, particulate and organic.

S(N,i,k,t) is the source of activity released to Region N of isotope I with physical form k.

L_{MN} represents the fractional leakage of activity from the region M to region N (fraction/sec). Leakage rates may be time dependent.

$F_{M,N}(k)$ represents the fraction of the activity leaving region M and flowing to region N, not filtered out by an intervening filter. Filter efficiency is a function of chemical form, k.

$F_{MN} = (1-FF_{MN})$, which represents the material that is not filtered, where FF_{MN} is the filter efficiency for flow of material from region M to region N (unitless).

$\lambda_d(I)$ - is the isotope decay constant (per second).

λ_s (N,k) - is the removal constant in region N (due to natural processes or sprays, if available) (per second). This parameter may vary with time.

The activity release is divided into the following isotope classes:

- Iodine
- Cesium
- Tellurium
- Noble Gases
- Other

Iodine may be distributed among three physical forms (as described above). Cesium and Other are particulate. Tellurium is treated as elemental iodine for purposes of activity transport. Noble gases are treated as a gas. Noble gas and organic iodine are not subject to settling or removal via plate-out/deposition processes.

Region 1 (Environment)

Region 1 calculations are not performed.

Region 2 (Drywell)

$$\frac{dA(2,i,k)}{dt} = -\lambda_d(i)A(2,i,k) - \lambda_s(2,k)A(2,i,k) + S(2,i,k)$$
$$+ L_{32}A(3,i,k) - (L_{23} + L_{27} + L_{25} + L_{21})A(2,i,k)$$

Where:

The source term S(2,i,k) is introduced based on the revised source term. This model allows linear introduction of mass from the reactor vessel at differing rates over different intervals.

S(2,i,k) is introduced according to the following equation:

$$S(2,i,k,t) = A_0(i)EXP(-\lambda_d(i)t) * (F(i,t))$$

Where:

$A_0(I)$ is the initial activity, of isotope I, in the core and F(i,t) is the incremental fraction of inventory of isotope I added to Region 2 as a function of time.

Region 3 (Wetwell)

$$\frac{dA(3,i,k)}{dt} = -\lambda_d(i)A(3,i,k) - \lambda_s(3,k)A(3,i,k) - L_{32}A(3,i,k)$$
$$- [L_{31F} + L_{31U}]A(3,i,k) + L_{23}A(2,i,k)F_{23} - L_{35}A(3,i,k)$$

Where:

$$F_{23} = F_{pool} \quad t < T_{bd}$$
$$= 1 \quad t > T_{bd}$$

F_{23} is the filter efficiency of the wetwell (note that the filter efficiency accounts for the flow that bypasses the suppression pool through the drywell/wetwell vacuum breakers).

F_{pool} represents the pool scrubbing efficiency and T_{bd} is the time frame for which the scrubbing applies.

In the above equation, L_{23} refers to the flow from the drywell to the wetwell. Early in the blowdown, L_{23} passes through the bottom of the vent pipe and through the suppression pool where the radionuclides are filtered. Later in the transient, the vent pipe vacuum breakers open and the pressures in the drywell and wetwell equalize. At these times L_{23} mixes directly with the atmosphere of the wetwell. The reverse flow (L_{32}) is initially zero until the vacuum breakers open. At that time the volumetric flow for the paths is assumed to equalize and the atmospheres in the two regions undergo rapid turnover.

Region 4 (Control Room)

The control room is exposed to radioactive releases from elsewhere in the plant. Air into the control room can enter via an unfiltered and filtered pathway. Environmental releases that influence the control room dose include (see Figure 9):

a. Filtered releases from the SGTS (L_{51})

b. Releases from the stack room (L_{61})

c. Unfiltered releases from the hardened wetwell vent (L_{31U})

d. CAD (Containment Air Dilution) System releases from the wetwell (L_{31F})

e. Unfiltered releases from the main condenser (L_{81})

f. Unfiltered releases from the Drywell (L_{21}). Note that no unfiltered releases from the drywell are considered for three SGTS fans in operation.

$$\frac{dA(4,i,k)}{dt} = -\lambda_d(i)A(4,i,k) - \left(\frac{L_{14U}}{V_{CR}} + \frac{L_{14F}}{V_{CR}}\right)A(4,i,k)$$

$$+(\chi/Q)_{5,CR}(L_{14U}+F_{14}(k)L_{14F})A(5,i,k)F_{51}(k)L_{51}$$

$$+(\chi/Q)_{5,CR}(L_{14U}+F_{14}(k)L_{14F})A(3,i,k)F_{31}(k)L_{31F}(1-F_{fbyp})$$

$$+(\chi/Q)_{5,CR}(L_{14U}+F_{14}(k)L_{14F})A(3,i,k)L_{31U}$$

$$+(\chi/Q)_{6,CR}(L_{14U}+F_{14}(k)L_{14F})A(6,i,k)L_{61}+(\chi/Q)_{2,CR}(L_{14U}+F_{14}(k)L_{14F})A(2,i,k)L_{21}$$

$$+(\chi/Q)_{8,CR}(L_{14U}+F_{14}(k)L_{14F})A(8,i,k)L_{81}$$

In the above equation the following should be noted:

- Radionuclides entering the control room pass through the CREVS. In the process, L_{14U} represents the unfiltered inflow (bypasses the CREVS filter). L_{14F} represents the filtered portion of this inflow. Unlike the general case, these are volumetric rather than fractional flow rates.

- The activity entering the control room is assumed to mix with the control room volume (V_{CR}). The exhaust flow of activity represents that due to the ventilation flow , as diluted by the control room air volume.

- All releases that pass through Region 5 are given the χ/Q associated with the stack (χ/Q's are atmospheric dispersion factors, the values of χ/Q varies with time and location, i.e., control room, EAB, or low population zone). This includes L_{31U}, L_{31F} and L_{51}. All other releases are given dispersion coefficients associated with the particular region. Note that L_{31F} is reduced by F_{fbyp}, which is the fraction which bypasses the stack.

Region 5 (Reactor Building)

$$\frac{dA(5,i,k)}{dt} = -(\lambda_d(i)+L_{51}+L_{56})A(5,i,k)+(L_{25})A(2,I,K)+(L_{35})A(3,i,k)$$

The Reactor Building receives flow from the drywell and wetwell. Flow leaving the Reactor Building passes through the SGTS and is either directly released to the environment via the stack or leaked into the stack room.

Region 6 (Room at Base of Stack)

$$\frac{dA(6,i,k)}{dt} = -\lambda_d(i)A(6,i,k) + (L_{56})F_{51}(k)A(5,i,k) + L_{31F}(F_{31})A(3,I,K)F_{fbyp} - L_{61}A(6,i,k)$$

CAD flow is filtered by SGTS and released from the stack and from Region 6 (stackroom).

Region 7 (Main Steam Line Piping)

$$\frac{dA(7,i,k)}{dt} = -\lambda_d(i)A(7,i,k) + L_{27}A(2,i,k)\,F_{27}(t) - L_{78}A(7,i,k)$$

L_{27} refers to leakage from the drywell (Region 2) into the main steam line (Region 7) and into the main condenser (Region 8). F_{27} represents settling in the pipe due to natural processes.

Region 8 (Main Condenser)

$$\frac{dA(8,i,k)}{dt} = -\lambda_d(i)\,A(8,i,k) - \lambda_s(8,k)\,A(8,i,k) + L_{78}A(7,i,k) - L_{81}A(8,i,k)$$

$\lambda_s(8,k)$ represents settling in the main condenser (radionuclide removal from the gas space). However, no radionuclide removal in the condenser is considered in the analysis.

2. Dose Calculation Methodology

The Dose calculation methodology used in this analysis is based on the DCF (Dose Conversion Factor) concept. The breathing rates and occupancy factors, radionuclide decay constants and dose conversion factors, and the atmospheric dispersion factors are provided in Tables 3 through 5. The resultant dose equations are as follows.

a. Offsite Whole Body Dose resulting from exposure to isotope I:

$$D^i_{wb}(t,N,M) = DCF^i_{wb}(\chi(t)/Q)_N(Q(t)^i_M)$$

Where DCF^i_{wb} is the whole body dose conversion factor for isotope I {(rem-m^3)/(Ci-sec)}.

In the above equation $(\chi(t)/Q)_N$ is the time dependent atmospheric dispersion factor to offsite location N. This factor is in units of sec/cubic meter.

$Q(t)^i_M$ is the integrated activity of isotope I, released to the environment via leakage path M, over time span t in Ci.

b. Offsite Skin Dose

$$D^{i}_{skin}(t,N,M) = DCF^{i}_{skin}(\chi(t)/Q)_N (Q(t)^{i}_{M})$$

DCF_{skin} is analogous to DCF_{wb}

c. Offsite Thyroid Dose

$$D^{i}_{th}(t,N,M) = DCF^{i}_{Th}(\chi(t)/Q)_N \cdot B(t) \cdot (Q(t)^{i}_{M})$$

B(t) represents the time dependent breathing rate for an adult.

Dose conversion factors for the thyroid dose refers to inhalation. The Thyroid dose is applied to all iodides and Tellurium (which is treated as I-132).

d. Control Room Doses

Personnel doses within the control room result from the ingress of outside air which contains radioisotopes from the various leakage pathways.

The calculation of control room whole body doses are calculated based as follows:

$$D^{i}_{wbCR}(t,M) = (VCR)^{.338}\,\frac{DCF^{i}_{wb}CRO(t)\,IQ(t)^{i}_{M}}{(1173)\,VCR(.02832)}$$

Where, CRO(t) represents the control room occupancy as a function of time.

The factor $(VCR^{0.338})/_{1173}$ is a geometrical correction factor to ratio a finite cloud to an infinite cloud.

VCR is the control room volume in ft^3; IQ is the integrated activity in the control room over the time interval under study.

$$IQ(t)^{i}_{M}\int A(4,i,k,t)\,Dt$$

Skin Dose within the Control Room:

$$D^{i}_{skinCR}(t,M) = \frac{DCF^{i}_{skin}CRO(t)\,IQ(t)^{i}_{M}}{VCR(.02832)}$$

Where, D_{skinCR} is the dose to the skin from beta radiation from all sources within the control room.

Thyroid dose within the control room is based on the inhaled dose and involves the breathing rate, B(t):

$$D_{th}, CR^{i}(t, M) = \frac{(DCF^{i}_{th}(CRO(t))\, IQ(t)^{i}_{M}) B(t)}{VCR(.02832)}$$

Where, DCF is in units of Rem per Curie inhaled.

3. Solution Methodology of Radionuclide Transport and Dose Calculation

The equation solution methodology is as follows:

a. Determine dose parameters and perform units conversion

b. Establish flows , filter efficiencies, and dispersion factors for input into solution matrix.

c. Solve equations in matrix form. The equation to be solved is of the form:

$$\frac{DA(t,i,k)}{Dt} = K \cdot A(t,i,k) + F$$

where, K is the instantaneous coefficient matrix and F is a column vector with time dependent constants (sources, etc.).

d. The matrix equation is recast into an implicit time dependent solution algorithm.

e. Once the activities are established in the previous step, the activities are combined in various form to obtain whole body, skin and thyroid doses at the exclusion area boundary, low population zone and control room.

H. OPERATOR AND OFFSITE DOSES AND CONCLUSIONS

The criteria for control room habitability given in 10 CFR 50, Appendix A, Criterion 19 - Control Room and NUREG-0800 - Standard Review Plan, Section 6.4, and is as follows:

Limit for 30 day dose accumulation:

5 Rem whole body
30 Rem Thyroid (iodine inhalation)
30 Rem skin

The allowable offsite doses are given in 10 CFR 100.11 to be:

25 Rem total whole body
300 Rem total to the thyroid due to iodine exposure

These limits apply to EAB 2 hour and LPZ thirty day doses. The results of the dose calculations are summarized in the table below:

SUMMARY OF DOSE CALCULATION RESULTS

LOCATION	DOSE TYPE	2 HR DOSE (REM)	30 DAY DOSE (REM)
CR	THYROID		17.9
CR	SKIN		1.79
CR	WHOLE BODY		0.046
EAB	THYROID	3.16	
EAB	SKIN	0.0566	
EAB	WHOLE BODY	0.075	
LPZ	THYROID		5.79
LPZ	SKIN		0.493
LPZ	WHOLE BODY		0.282

The contribution of other isotopes to the whole body LPZ and EAB doses are as follows:

EAB 2 hour: 0.000527 Rem
LPZ 30 day: 0.00034 Rem

Iodine -131 contributes 16.64 Rem to the control room thyroid dose (30 day), distributed among the three iodine forms as follows:

Elemental: 3.21 Rem
Organic: 12.5 Rem
Particulate: 0.923 Rem

The control room personnel dose due to external exposure of the control room (from the reactor building and core spray piping) has not been recalculated in this analysis. The combined reactor building/ core spray piping external exposure doses were determined in a previous calculation to be 1.5 Rem over 30 days with about 1/3 of that total coming from the core spray piping. It is estimated that the core spray piping dose could increase by a factor of 1.5 using the NUREG-1465 source term (due entirely to the increased radiocesium which would make its contribution in the long-term when control room occupancy factors are less than unity), but there is expected to be a substantial corresponding decrease in the reactor building airborne contribution which accounts for 2/3 of the total dose. Moreover, with application of the revised source term the whole body dose contribution from sources within the control room has remained low (less than 0.5 Rem); and therefore, the 10 CFR 50, Appendix A, General Design Criterion 19 whole body dose acceptance value of 5 Rem will not be exceeded.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Browns Ferry Nuclear Plant (BFN) Units 1, 2 and 3 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. **The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

The proposed change increases the allowable leak rate specified for the main steam isolation valves (MSIVs) and reflects the exclusion of the measured MSIV leakage from the combined local leak rate test results. These changes do not affect the precursors for any accident or transient analyzed in Chapter 14 of the BFN Updated Final Safety Analysis Report (UFSAR). Therefore, there is no increase in the probability of any accident previously evaluated.

Plant specific radiological analyses have been performed to asses the effects of the proposed increase in the allowable MSIV leakage rate and reflects the exclusion of the measured MSIV leakage from the combined local leak rate test results. The resulting changes in containment leakage and the revised post-accident source term (NUREG-1465) have been analyzed in terms of control room, Technical Support Center, and offsite doses following the worst case design basis accident (a double ended guillotine recirculation line break induced loss of coolant accident [LOCA]). The radiological analyses used conservative assumptions. The analyses demonstrated that the resulting doses were below the regulatory limits contained in 10 CFR 100, Reactor Site Criteria, and 10 CFR 50, Appendix A, General Design Criterion 19, Control Room. Therefore, the proposed changes do not involve a significant increase in the consequences of any accident previously evaluated.

B. **The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.**

The proposed increase in the allowable MSIV leak rate and the exclusion of the measured MSIV leakage from the combined local leak rate test results does not reflect a modification to the MSIVs that could impact their ability to isolate primary containment. No new failure modes or potential for operator errors are introduced.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **The proposed amendment does not involve a significant reduction in a margin of safety.**

Plant specific radiological analyses have been performed to asses the effects of the proposed increase in the allowable MSIV leakage rate and the exclusion of the measured MSIV leakage from the combined local leak rate test results. The analyses evaluated the control room, Technical Support Center, and offsite doses following the worst case design basis accident (a double ended guillotine recirculation line break induced loss of coolant accident [LOCA]). The radiological analyses used conservative assumptions. The application of the revised source term should be judged by the same licensing acceptance limits in use with the existing source term. The revised BFN design basis (i.e., revised source term coupled with existing regulatory guidance for the calculation of post-design basis accident consequences) demonstrated that the resulting doses were below the regulatory limits contained in 10 CFR 100, and 10 CFR 50, Appendix A, General Design Criterion 19. An acceptable margin of safety is inherent in these licensing acceptance limits, and the improvement in the technical knowledge base and in the analytical techniques that are part of the revised post-accident source term and the modeling of the Main Steam Isolation Valve (MSIV) leakages does not alter the acceptability of the margin. Therefore, the resulting calculated doses, which are below regulatory limits, assure that the proposed amendment does not involve a significant reduction in the margin of safety.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

TABLE 1
RELEVANT DIMENSIONAL INFORMATION FOR BUILDINGS, AIR INTAKES, AND VENT LOCATIONS

General information about the building dimensions and relevant dimensions are provided below:

Reactor Building -

- Height above plant grade: 152 feet (46.3 meters)
- Length [North-South]: 119 feet (36.3 meters)
- Width [East-West]: 468 feet (143 meters)

Control Building -

- Height above plant grade: 69.6 feet (21.2 meters)
- Length [North-South]: 41 feet (12.5 meters)
- Width [East-West]: 465 feet (142 meters)

Turbine Building -

- Height above plant grade: 117 feet (35.7 meters)
- Length [North-South]: 345 feet (105 meters)
- Width [East-West]: 420 feet (128 meters)

- Location of intakes for CREVS:
  - Height above plant grade: 69 feet (21 meters)
  - The Unit 1 intake is located 6.25 feet (1.91 meters) West of the Turbine Building West wall
  - The Unit 3 intake is located 6.25 feet (1.91 meters) East of the Turbine Building East wall

- Location of Turbine Building Vents:
  - The Turbine Building vents (exhaust fans) are located in three groups. Each group consists of 9 fans. The Unit 1 group of fans is located 97.9 feet (29.9 meters) from the West face Turbine Building wall. The Unit 2 group of fans is located 165 feet (50.3 meters) from the West face Turbine Building wall. The Unit 3 group of fans is located 322 feet (98.1 meters) from the West face Turbine Building wall. All the fans are located 117 feet (35.7 meters) above plant grade.

TABLE 1
DIMENSIONAL INPUTS USED
IN CONTROL ROOM OPERATOR DOSE CALCULATIONS
(CONTINUED)

- Distance Between Intakes for CREVS and Plant Stack:

  - Unit 1 intake is 545 feet (166 meters)
  - Unit 3 intake is 928 feet (283 meters)

- Distance Between Nearest Turbine Building Vents and Intakes for CREVS:

  - Unit 1 fan group to Unit 1 intake is 106 feet (32.3 meters)
  - Unit 2 fan group to Unit 1 intake is 172 feet (52.4 meters)
  - Unit 3 fan group to Unit 1 intake is 328 feet (100 meters)
  - Unit 1 fan group to Unit 3 intake is 328 feet (100 meters)
  - Unit 2 fan group to Unit 3 intake is 262 feet (79.8 meters)
  - Unit 3 fan group to Unit 3 intake is 105 feet (32.0 meters)

Plant Stack -

Height above plant grade: (602 feet) (183 meters)
Base - outside diameter : (62.4 feet) (19.0 meters)
Top - outside diameter: (6.1 feet) (1.8 meters)

The inside diameter of the stack at the top is 3 ft. 4¾ in. The top of the stack is 534 ft. above the control bay air intakes.

TABLE 2
TOTAL STORED ENERGY

Interval Start Sec	Radionuclide	Initial Source Strength Mev/sec-Mw	Decay Constant-sec^{-1}	Strength at Interval Start Mev/sec-Mw	0.5(Decay Constant)$^{-1}$x (1-$e^{-\text{Decay Constant x 5400 seconds}}$)x Source Strength - Mev/Mw
1830	I-131	3.63E14 x 0.7	9.96E-7	2.54E14	6.84E17
	I-132	2.82E15 x 0.7	8.27E-5	1.70E15	3.70E18
	I-133	1.15E15 x 0.7	9.22E-6	7.92E14	2.09E18
	I-134	3.10E15 x 0.7	2.23E-4	1.44E15	2.26E18
	I-135	2.90E15 x 0.7	2.86E-5	1.93E15	4.83E18
	Other solids	3.72E16	7.05E-5	3.27E16	7.34E19
	Total	4.44E16		3.88E16	8.70E19

The total stored energy = o.70E19 Mev/Mw = 4.57E7 BTU for a core power of 3458 Mw.

TABLE 3
BREATHING RATES AND OCCUPANCY FACTORS

BREATHING RATES

END TIME (HR)	RATE (M^3/SEC)
8	3.47E-04
24	1.75E-04
720	2.32E-04

OCCUPANCY FACTORS

END TIME (HRS)	FACTOR
24	1
96	0.6
720	0.4

TABLE 4
RADIONUCLIDE DECAY CONSTANTS AND DOSE CONVERSION FACTORS

ISOTOPE	DECAY LAMBDA (DIS /SEC)	WHOLE BODY DCF Rem-m^3/(Ci-S)	BETA DCF (SKIN) Rem-m^3/(Ci-S)	THYROID DCF (10^4/Ci*)
Kr-83m	1.04E-04	1.27E-05	0	0
Kr-85m	4.39E-05	2.30E-02	4.97E-02	0
Kr-85	2.04E-09	3.31E-04	4.84E-02	0
Kr-87	1.52E-04	1.33E-01	3.36E-01	0
Kr-88	6.89E-05	3.38E-01	7.76E-02	0
Kr-89	3.63E-03	3.03E-01	3.47E-01	0
Kr-90**	.215E-1	0	0	0
Xe-131m	6.68E-07	1.25E-03	1.33E-02	0
Xe-133m	3.49E-06	4.29E-03	2.96E-02	0
Xe-133	1.52E-06	4.96E-03	9.67E-03	0
Xe-135m	7.40E-04	6.37E-02	2.14E-02	0
Xe-135	2.09E-05	3.59E-02	6.32E-02	0
Xe-137	2.96E-03	2.83E-02	4.59E-01	0
Xe-138	6.80E-04	1.87E-01	1.47E-01	0
I-131	9.96E-07	5.59E-02	3.07E-02	110
I-132	8.27E-05	3.55E-01	1.10E-01	0.63
I-133	9.22E-06	9.11E-02	8.90E-02	18
I-134	2.23E-04	4.11E-01	1.42E-01	0.11
I-135	2.86E-05	2.49E-01	7.86E-02	3.1
Cs-134	9.55E-09	2.58E-01	1.15E-01	0
Cs-137	7.29E-10	9.30E-02	1.27E-01	0
Te-132	2.51E-06	3.55E-01	1.10E-01	0.63
Other	7.05E-5	.168	0	0

* INHALED   ** DCF set = 0.00

TABLE 5
ATMOSPHERIC DISPERSION FACTORS (X/Qs)(SEC/M^3)
SHEET 1 OF 2

NODE 5 - STACK RELEASE

TIME INTERVAL (HRS)	EAB (X/Q)	LPZ (X/Q)	CR (X/Q)
0 TO 1.5	9.70E-07	8.00E-07	5.91E-15
1.5 TO 2	2.40E-05	1.300E-05	3.31E-15
2 TO 8		8.00E-07	3.80E-15
8 TO 24		4.00E-07	3.00E-15
24 TO 96		2.00E-07	1.90E-15
96 TO 720		6.50E-08	9.60E-16

NODE 6 - STACK ROOM RELEASE

TIME INTERVAL (HRS)	EAB (X/Q)	LPZ (X/Q)	CR (X/Q)
0 TO 2	1.22E-04	5.65E-05	8.89E-04
2 TO 8		5.65E-05	7.30E-04
8 TO 24		2.24E-05	6.60E-04
24 TO 96		7.94E-06	5.40E-04
96 TO 720		1.71E-06	4.00E-04

TABLE 5
ATMOSPHERIC DISPERSION FACTORS (X/Qs)(SEC/M^3)
SHEET 2 OF 2

NODE 8 - MAIN CONDENSER RELEASE

TIME INTERVAL (HRS)	EAB (X/Q)	LPZ (X/Q)	CR (X/Q)
0 TO 2	2.70E-04	1.32E-04	1.74E-04
2 TO 8		6.02E-05	1.47E-04
8 TO 24		4.07E-05	1.27E-04
24 TO 96		1.73E-05	1.01E-04
96 TO 720		5.10E-06	7.20E-05

NOTES: EAB - Exclusion Area Boundary
LPZ - Low Population Zone
CR - Control Room

The fumigation time interval is selected at the worst one half hour period over the first two hours. This occurs in the 1.5 to 2 hour time frame. Including this effect later in the event acknowledges the impact of the later release of radionuclides. Doses will be maximized with this assumption since the later interval has the higher atmospheric releases.

FIGURE 1
GENERAL PLANT LAYOUT

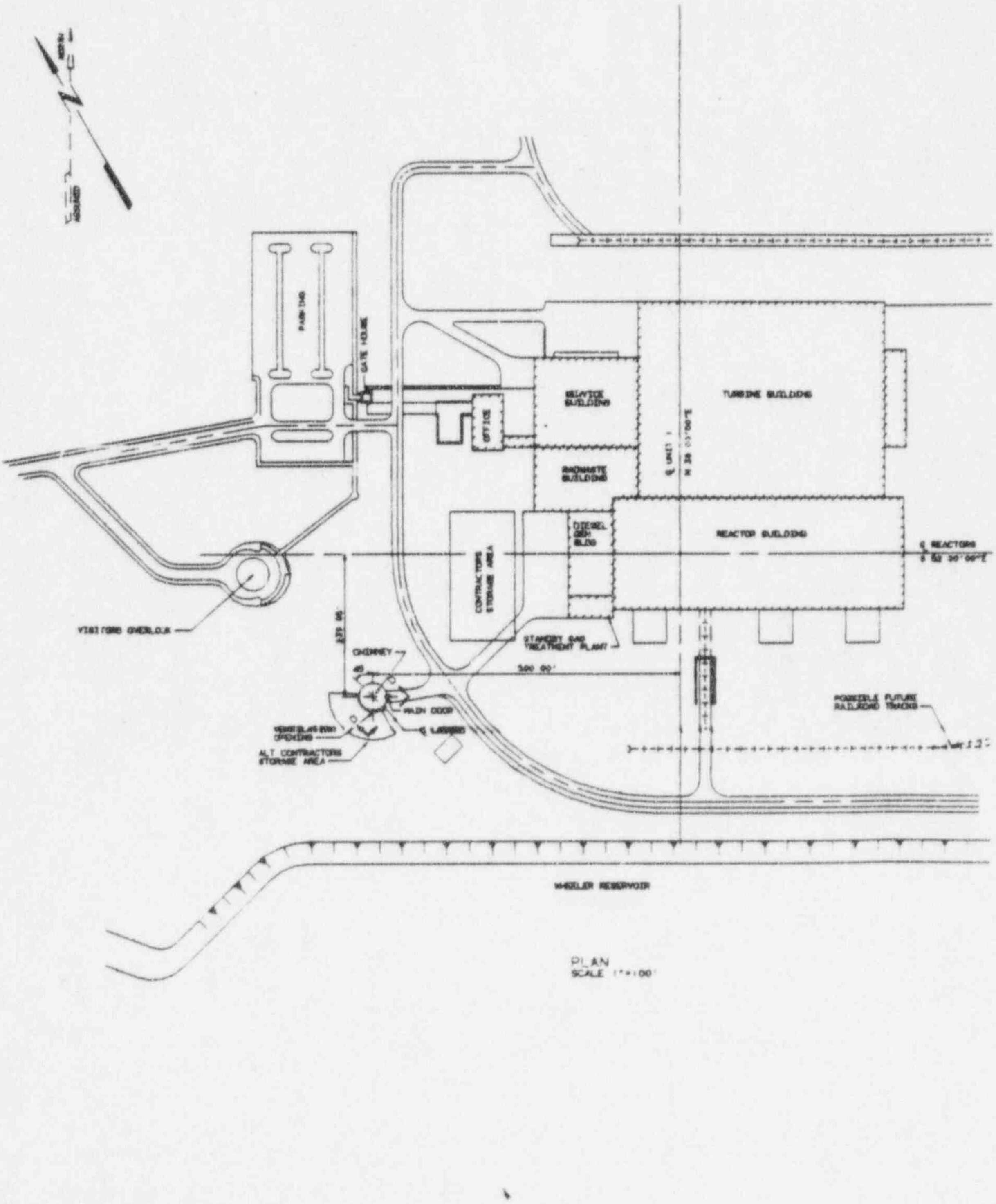


FIGURE 2
OVERALL SITE LAYOUT

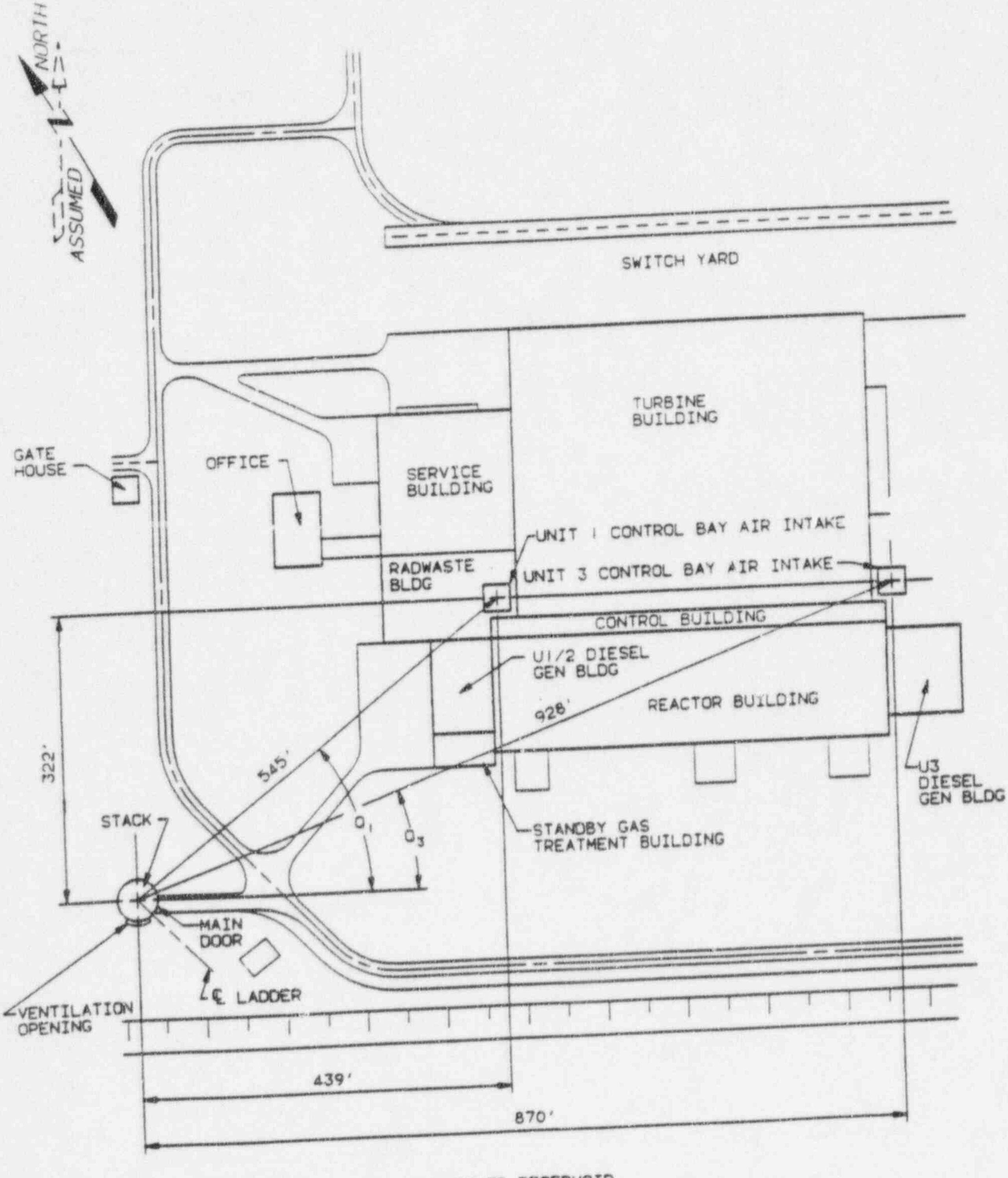


FIGURE 3
TRANSVERSE SECTION OF REACTOR, CONTROL AND TURBINE BUILDINGS

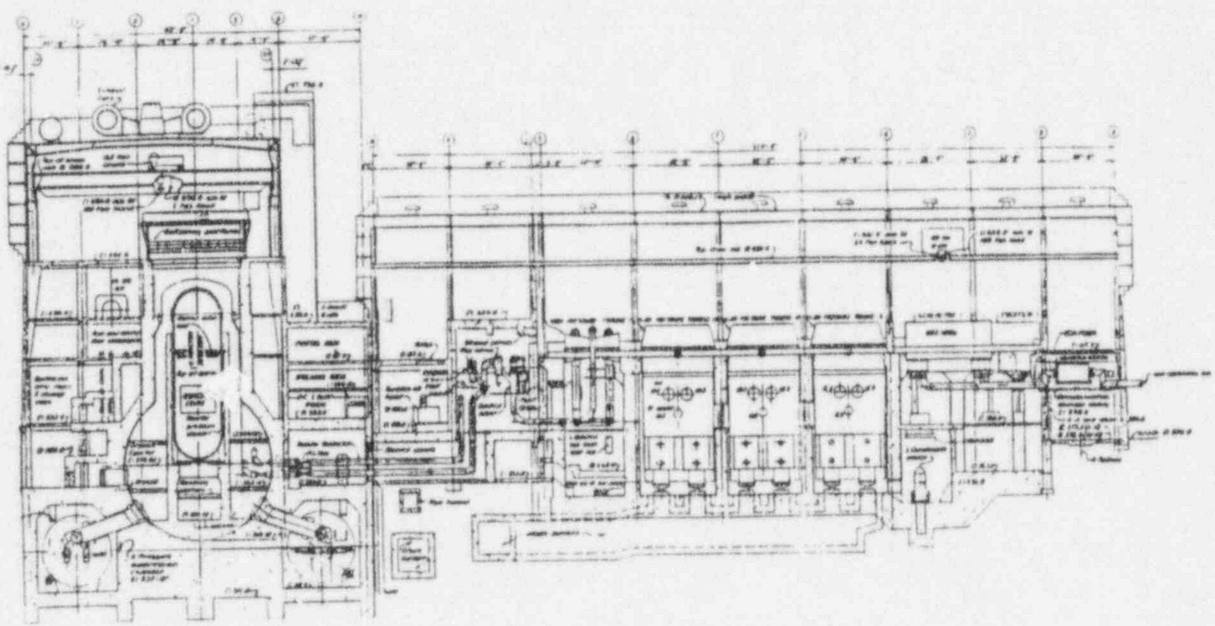


FIGURE 4
CONTROL ROOM EMERGENCY VENTILATION SYSTEM AIR INTAKES

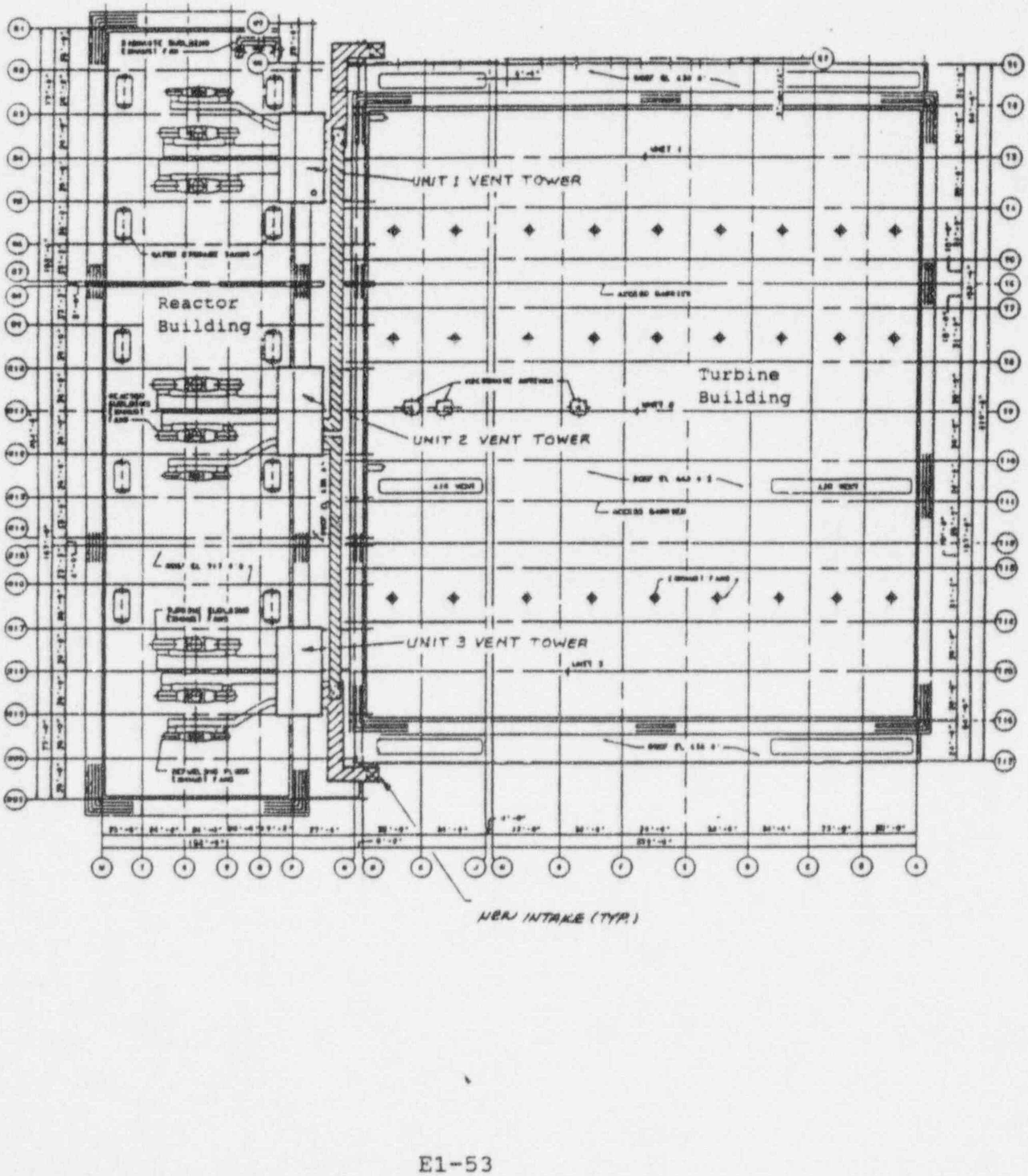
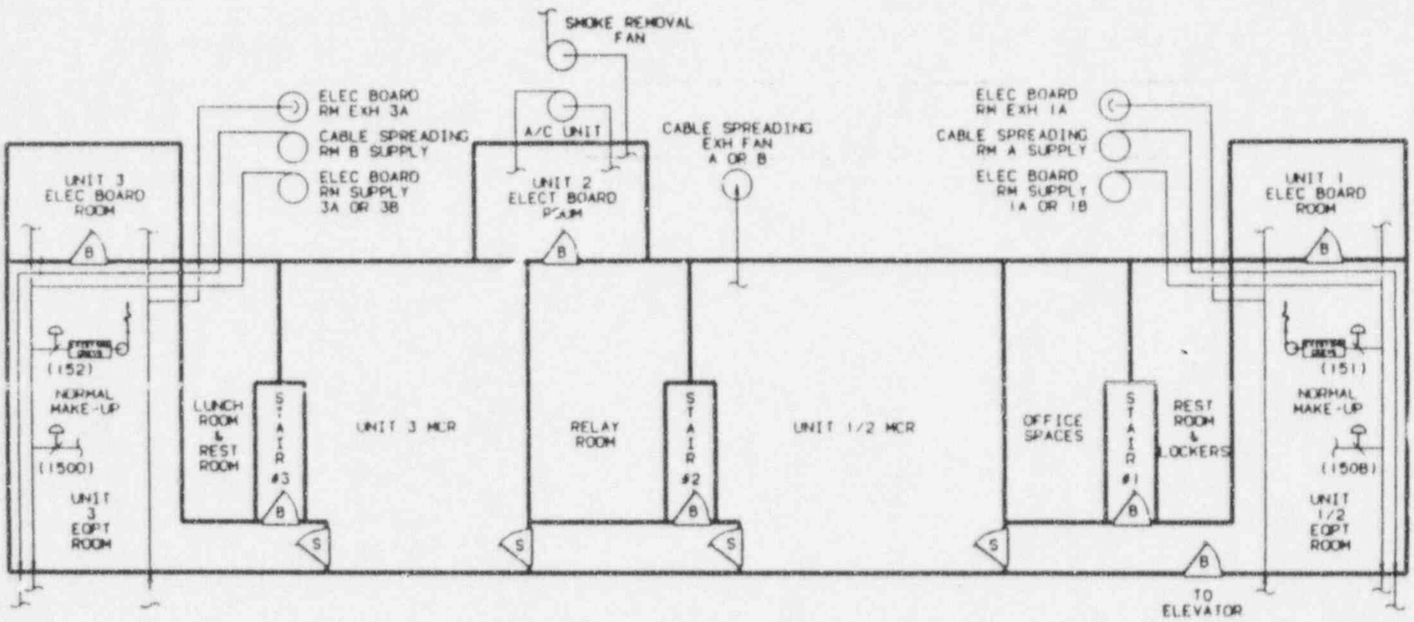


FIGURE 5
CONTROL BUILDING HEATING, VENTILATING, AND AIR CONDITIONING (HVAC) SYSTEM



EL 617 CONTROL BUILDING
HVAC EQUIPMENT SCHEMATIC

LEGEND:
B = BOUNDARY DOORS (7)
S = SECURITY DOORS (4)

FIGURE 6
POST-DESIGN BASIS ACCIDENT CONTAINMENT TEMPERATURE AND PRESSURE
SHEET 1 OF 2

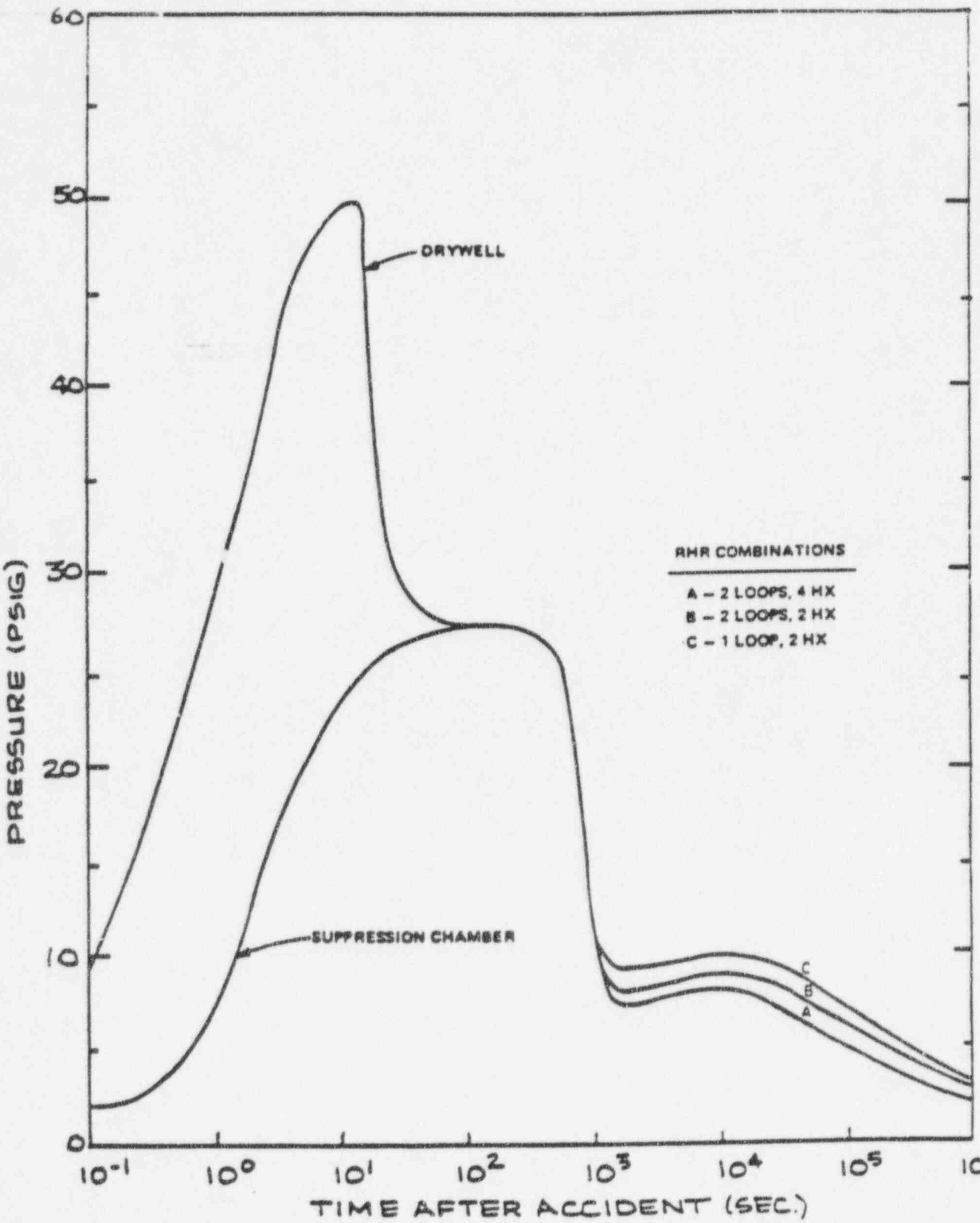


FIGURE 6
POST-DESIGN BASIS ACCIDENT CONTAINMENT TEMPERATURE AND PRESSURE
SHEET 2 OF 2

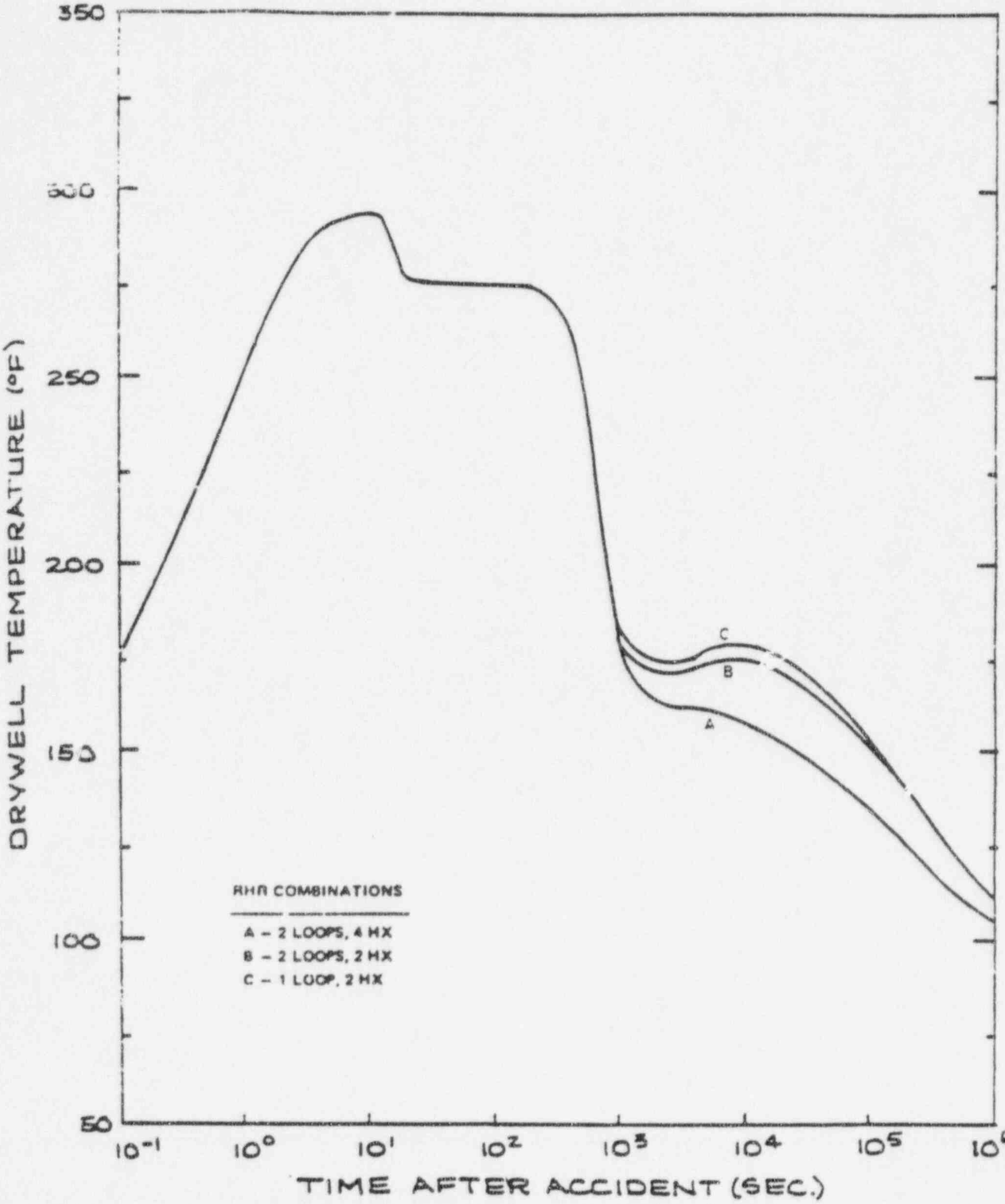
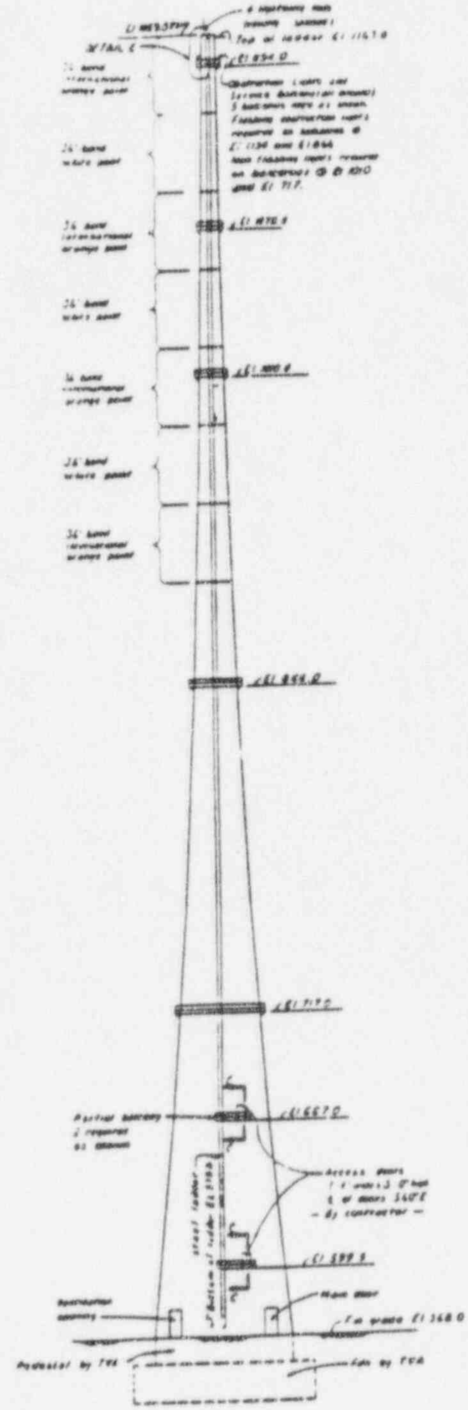


FIGURE 7
PLANT STACK



ELEVATION D-D

FIGURE 8
CALCULATIONAL MODEL NODAL ARRANGEMENT

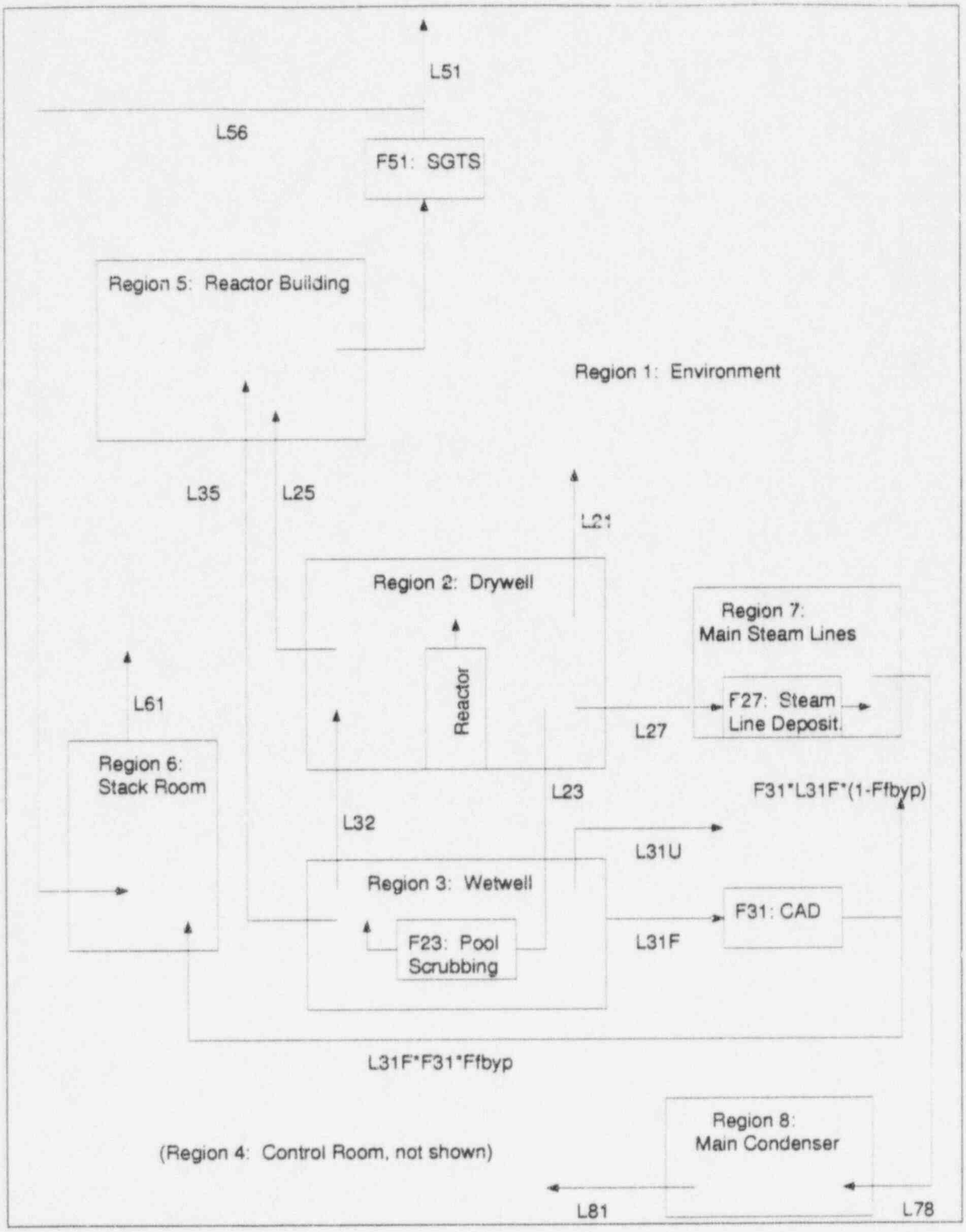


FIGURE 9
SOURCES OF ACTIVITY TO THE CONTROL ROOM

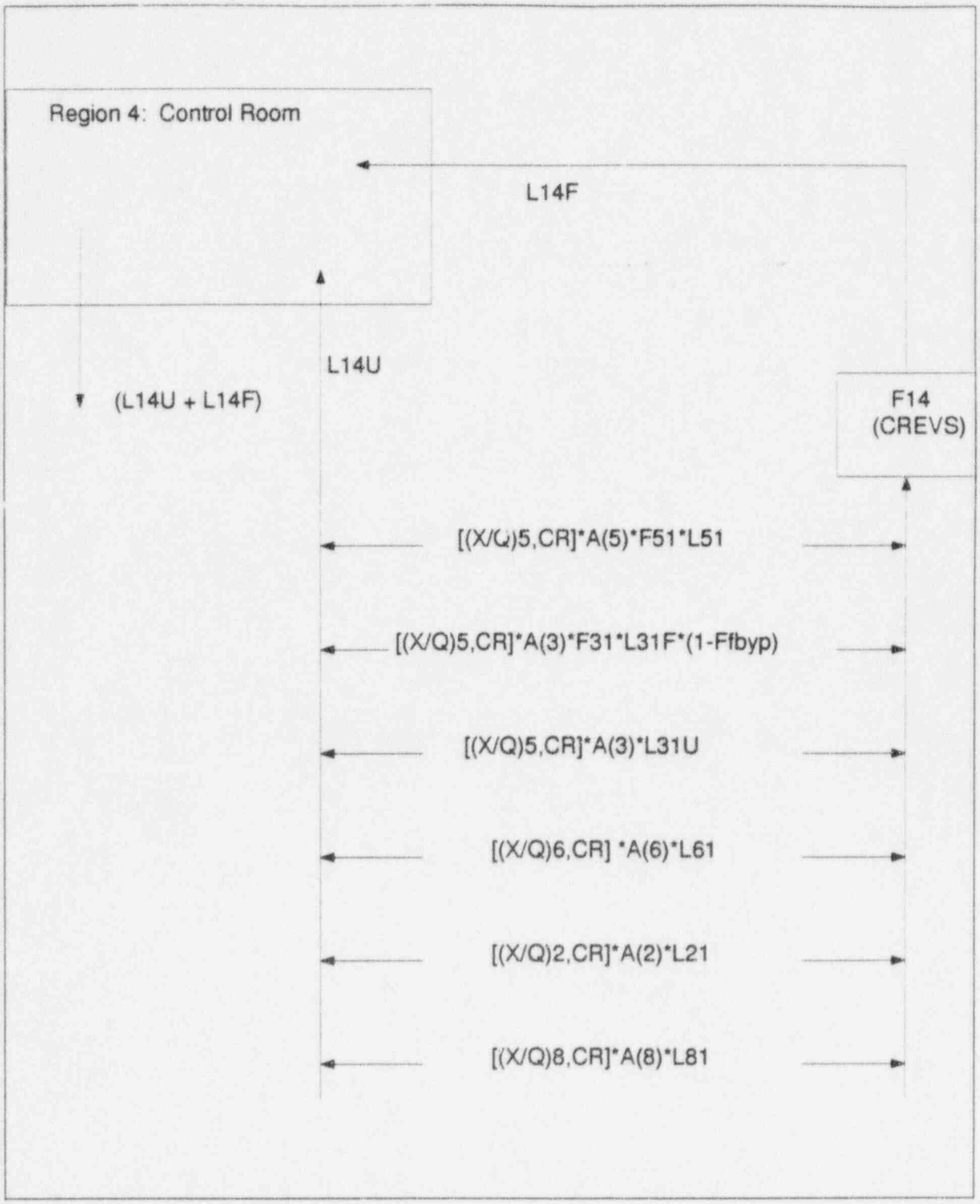


FIGURE 10
SOURCES OF ACTIVITY TO THE EXCLUSION AREA BOUNDARY

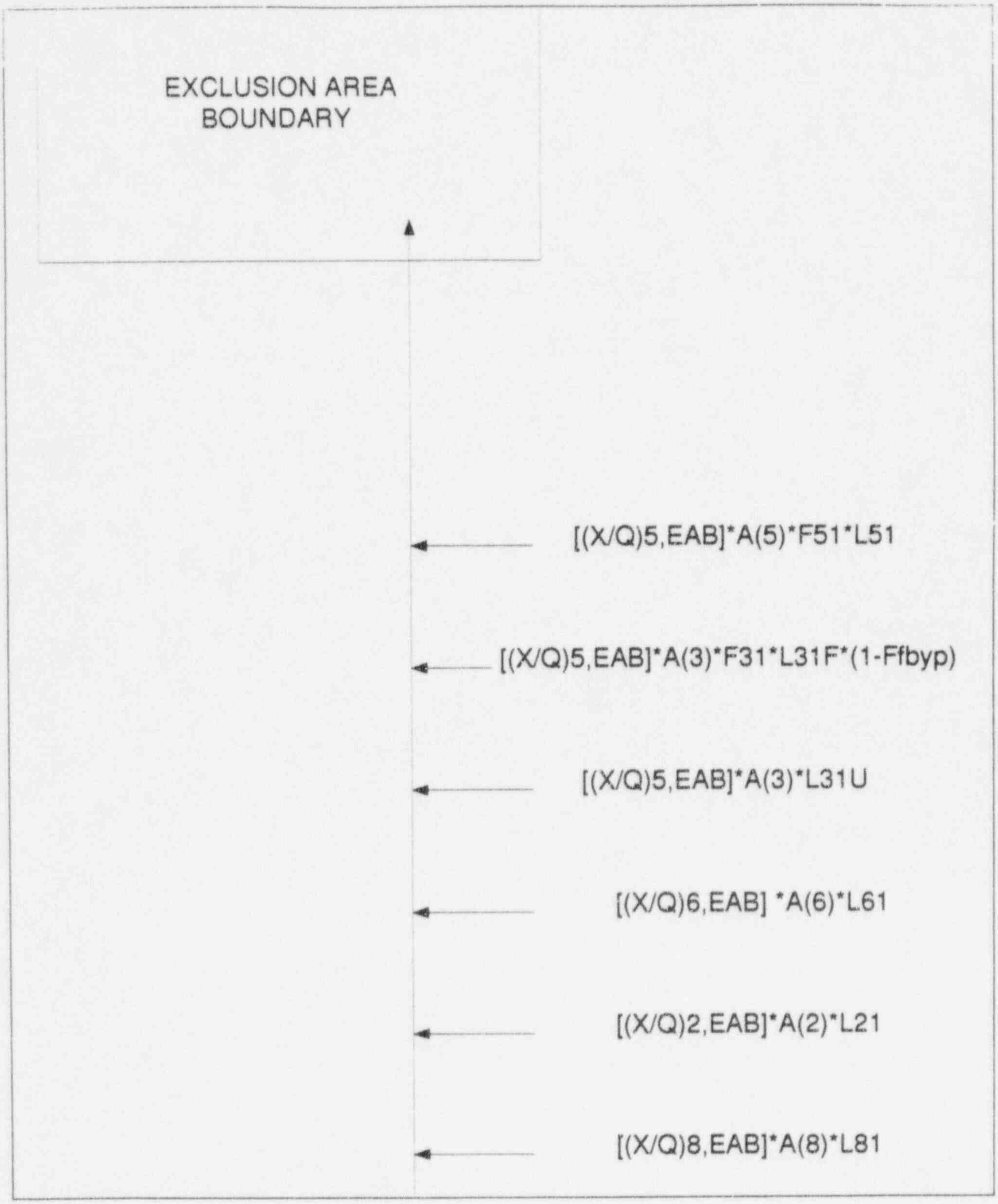
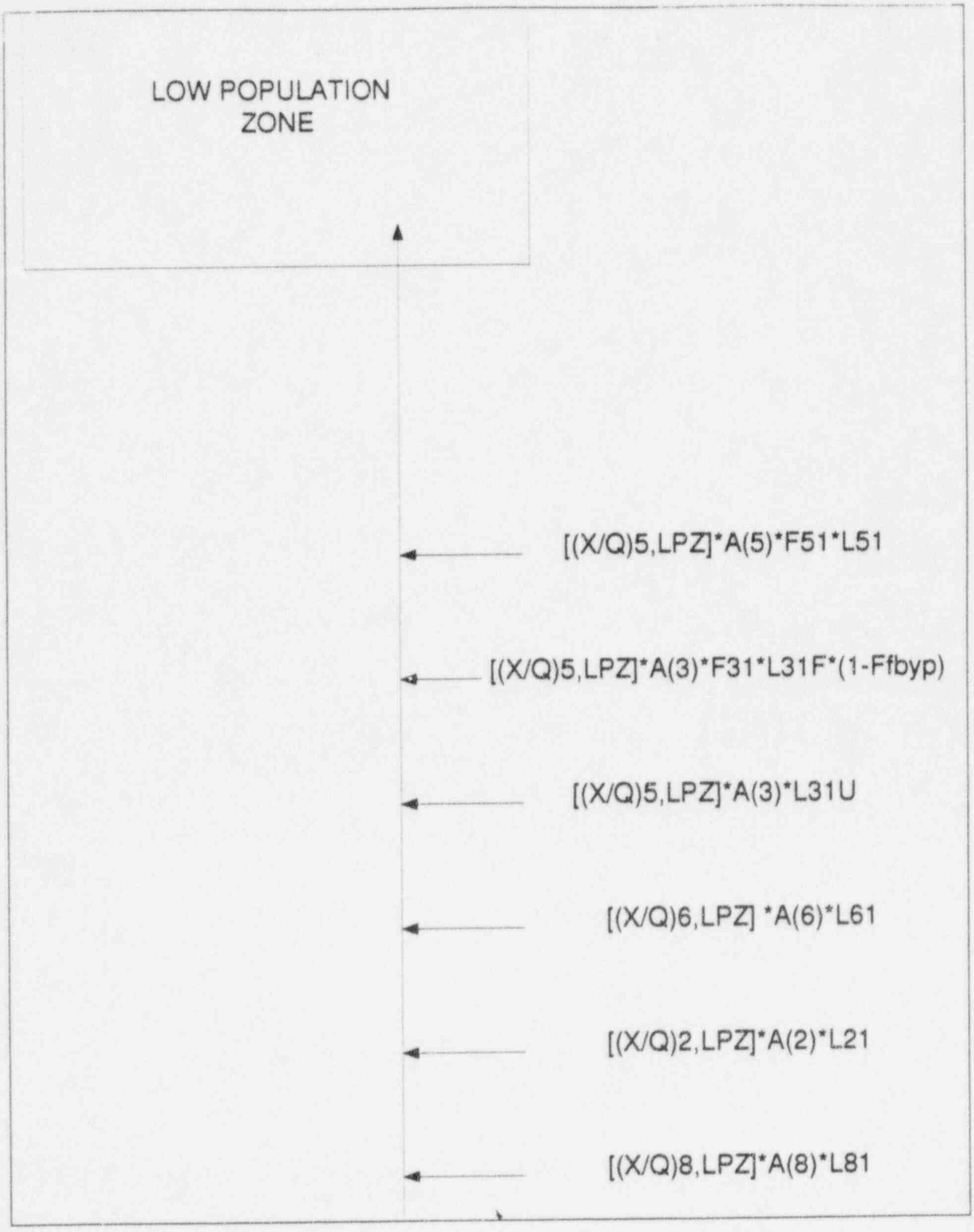


FIGURE 11
SOURCES OF ACTIVITY TO THE LOW POPULATION ZONE



ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-356
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
3.7/4.7-7	3.7/4.7-7	3.7/4.7-7
3.7/4.7-8	3.7/4.7-8	3.7/4.7-8

II. MARKED PAGES

See attached.

LIMITING CONDITIONS FOR OPERATION

, except the main steam
isolation valves,

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of ~~11.5 scf/hr for any one main steamline isolation valve~~ is exceeded, repairs and retest shall be performed to correct the condition.

100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines

LIMITING CONDITIONS FOR OPERATION

, except the main steam isolation valves,

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

The total path leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION

100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines

SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of ~~11.5 scf/hr for any one main steamline isolation valve~~ is exceeded, repairs and retest shall be performed to correct the condition.

NOV 16 1992

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
, except the main steam isolation valves,	4.7.A. Primary Containment 4.7.A.2.g (Cont'd) The total leakage from all penetrations and isolation valves shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of ~~11.5 scf/hr for any one main steamline isolation valve~~ is exceeded, repairs and retest shall be performed to correct the condition.

100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 scf/hr through all four main steam lines

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE TS-356
REVISED PAGES

I. AFFECTED PAGE LIST

Unit 1	Unit 2	Unit 3
3.7/4.7-7	3.7/4.7-7	3.7/4.7-7
3.7/4.7-8	3.7/4.7-8	3.7/4.7-8

II. REVISED PAGES

See attached.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2.g (Cont'd) The total leakage from all penetrations and isolation valves, except the main steam isolation valves, shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2. (Cont'd) h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately. (2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest. i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 sch/hr through all four main steam lines is exceeded, repairs and retest shall be performed to correct the condition.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment 4.7.A.2.g (Cont'd) The total path leakage from all penetrations and isolation valves, except the main steam isolation valves, shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 100 scf/hr for any one main steamline isolation valve or a total maximum pathway leakage rate of 250 sch/hr through all four main steam lines is exceeded, repairs and retest shall be performed to correct the condition.

LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS

4.7.A. Primary Containment

4.7.A.2.g (Cont'd)

The total leakage from all penetrations and isolation valves, except the main steam isolation valves, shall not exceed 60 percent of L_a per 24 hours. Leakage from containment isolation valves that terminate below suppression pool water level may be excluded from the total leakage provided a sufficient fluid inventory is available to ensure the sealing function for at least 30 days at a pressure of 54.6 psig. Leakage from containment isolation valves that are in closed-loop, seismic class I lines that will be water sealed during a DBA will be measured but will be excluded when computing the total leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
	4.7.A. Primary Containment

4.7.A.2. (Cont'd)

h. (1) If at any time it is determined that the criterion of 4.7.A.2.g is exceeded, repairs shall be initiated immediately.

(2) If conformance to the criterion of 4.7.A.2.g is nct demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

i. The main steamline isolation valves shall be tested at a pressure of 25 psig for leakage during each refueling outage. If the leakage rate of 100 scf/hr for any one main steamline isolation valve or a totai maximum pathway leakage rate of 250 sch/hr through all four main steam lines is exceeded, repairs and retest shall be performed to correct the condition.

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

EXEMPTION FROM 10 CFR 50, APPENDIX J, OPTION A,
SECTIONS II.H.4, III.C.2(a), AND III.C.3,
AND 10 CFR 50, APPENDIX J, OPTION B

I. APPLICABLE RULE

The requirements of 10 CFR 50, Appendix J, Option A, Sections II.H.4, III.C.2(a), and III.C.3 are as follows:

10 CFR 50, Appendix J, Section II.H.4 -

H. "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

10 CFR 50, Appendix J, Section III.C.3 -

Acceptance criterion. The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 L_a.

The definition of overall integrated leakage rate in 10 CFR 50, Appendix J, Option B, is as follows:

"Overall integrated leakage rate means the total leakage rate through all tested leakage paths, including containment welds, valves, fittings, and components that penetrate the containment system."

II. REQUESTED EXEMPTION

TVA requests an exemption from the requirements of 10 CFR 50, Appendix J, Sections II.H.4 and III.C.3 and 10 CFR 50, Appendix J, Option B. The exemption would allow the exclusion of the measured Main Steam Isolation Valve (MSIV) leakage from the combined local leak rate test results.

III. BACKGROUND

As discussed in Enclosure 1, the current Technical Specification allowable MSIV leakage rate (11.5 scf/hr) is extremely small considering the valve's physical size and operating characteristics (large size and fast-acting). Additionally, the ability of the turbine building equipment to contain the radioactive material was not considered at the time the leakage limit was established. Based on the in-depth evaluation of MSIV leakages, the Boiling Water Reactor Owners' Group (BWROG) has concluded that leakage rates of over 500 scf/hr are not indicative of substantial mechanical defects in the valves which would challenge the capability of the valves to fulfill their safety function of isolating the steam lines.

In addition, the NRC began a major research effort about 1981 to obtain a better understanding of fission-product transport and release mechanisms in light water reactors under severe accident conditions. This research effort has included extensive NRC staff and contractor efforts involving a number of national laboratories as well as nuclear industry groups. The current effort to revise the design basis accident source term started in 1990. This effort arose out of the technical initiative of the Advanced Light Water Reactor Program. Using this information, a revised accident source term has been developed (NUREG-1465, Accident Source Terms for Light-Water Nuclear Power Plants). The revised source term is expressed in terms of times and rates of appearance of radioactive fission products into the containment, the types and quantities of the species released, and other important attributes such as the chemical forms of iodine. This mechanistic approach presents a more realistic, but still conservative, portrayal of the amount of fission products present in the containment from a postulated severe accident. These assumptions have a significant affect on the design of engineered safety features.

These two efforts, the BWROG investigation of MSIV leakage rates and the development of a revised source term, have led TVA to perform a comprehensive evaluation of the consequences of design basis accidents, including the allowable leakage and surveillance testing of MSIVs. Based on this reevaluation, TVA has determined these exemptions to the requirements of 10 CFR 50, Appendix J, are prudent and justified.

IV. TECHNICAL EVALUATION

The function of the primary containment is to isolate and contain fission products released from the reactor primary system following a design basis accident and to confine the postulated release of radioactive material. The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate. Periodic testing of the leak tightness of the primary containment as well as individual penetrations and valves is necessary to assure that the assumed release rate in the plants' safety analysis is conservative.

Also as discussed in Enclosure 1, plant specific radiological analyses have been performed to asses the effects of the proposed increase in the allowable MSIV leakage rate in terms of control room, Technical Support Center, and offsite doses following the worst case design basis accident (a double ended guillotine recirculation line break induced loss of coolant accident [LOCA]). The radiological analyses used the revised accident source term for light-water nuclear power plants contained in NUREG-1465 and conservative assumptions. The contribution from the increased allowable MSIV leakage rate to the control room operator and offsite dose has been calculated separately. The analyses demonstrated that the resulting doses were below the regulatory limits contained in 10 CFR 100, Reactor Site Criteria, and 10 CFR 50, Appendix A, General Design Criterion 19, Control Room. Therefore, the exemptions to the requirements of 10 CFR 50, Appendix J, are acceptable and do not compromise the safety design basis of the primary containment or the overall purpose of performing leak rate testing.

V. JUSTIFICATION FOR EXEMPTION

10 CFR 50.12 authorizes the NRC to grant exemptions from its own requirements. An exemption must (1) be authorized by law, (2) not present an undue risk to the public health and safety, (3) be consistent with the common defense and security, and (4) must entail special circumstances.

1. AUTHORIZED BY LAW

   TVA was issued its Operating Licenses for BFN under the provisions of Section 104.b of the Atomic Energy Act. Operating Licenses issued by the Commission pursuant to Section 104.b are not limited, by statute, to specific methods of testing primary containment integrity. Thus, the Commission can legally exempt TVA from the requirements of 10 CFR 50, Appendix J.

2. NOT PRESENT AN UNDUE RISK TO THE PUBLIC HEALTH AND SAFETY

   The revised MSIV leakage rate has been incorporated in the radiological analysis for a postulated LOCA as an addition to the designed containment leak rate. The analysis demonstrates an acceptable increase to the dose exposures previously calculated for the control room and off-site. The revised LOCA doses remain well within the guidelines of 10 CFR 100 for off-site doses and 10 CFR 50, Appendix A, General Design Criterion 19, for the control room doses.

   In addition, Technical Specification Surveillance Requirement 4.7.A.2.i has provided for allowable MSIV leak rates, which assure that the MSIVs isolation function is not compromised. Finally, potential MSIV leakage is subjected to plate-out, and hold-up in the main steam piping and condenser, thus minimizing their effect on the total dose released.

Furthermore, the risk to the public health and safety will be reduced with the implementation of the proposed MSIV leakage treatment method. The implementation will provide BFN with a capability to process leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage rate limit. From a safety perspective, the proposed changes result in an increase in protection to the public. Therefore, the proposed exemption presents no undue risk to public health and safety.

3. BE CONSISTENT WITH THE COMMON DEFENSE AND SECURITY

The Commission's Statement of Considerations in support of the exemption rule note with approval the explanation of this standard as set forth in Long Island Lighting Company (Shoreham Nuclear Power Station, Unit 1), LBP-84-45, 20 NRC 1343, 1400 (October 29, 1984). There, the term "common defense and security" refers principally to the safeguarding of special nuclear material, the absence of foreign control over the applicant, the protection of Restricted Data, and the availability of special nuclear material for defense needs. The granting of the requested exemption will not affect any of these matters and, thus, such grants are consistent with the common defense and security.

4. MUST ENTAIL SPECIAL CIRCUMSTANCES

According to NRC regulations, special circumstances are present if any one of the six different cases cited in 10 CFR 50.12(a)(2) are present. TVA submits that the existence of special circumstances (ii), (iii) and (vi) are applicable for this exemption request:

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of the rule is to limit releases to within the off-site and control room dose guidelines of 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion 19. Compliance with Appendix J of 10 CFR 50 for Type A test acceptance criteria is not necessary to achieve the underlying purpose of the rule because MSIV leakage is not directed into the reactor primary containment. Instead, the MSIV's leakage is directed through the main steam drain piping into the condenser. Since Type A tests are intended to measure the primary containment overall integrated leak rate (ILRT), the MSIV's leakage rate should not be included in the measurement of the ILRT. Compliance with Appendix J of 10 CFR 50 Type C test acceptance criteria is not necessary because a specific MSIV leak rate limit is already specified in Technical Specifications Surveillance Requirement 4.7.A.2.i. The safety analysis has been revised to assess the radiological consequences of MSIV leakage following a design basis LOCA. The analysis has demonstrated that the revised LOCA doses are well within the off-site and control room dose guidelines of 10 CFR 100 and General Design Criterion 19.

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

Strict compliance with Appendix A of 10 CFR 50 Type A and Type C test acceptance criteria results in undue hardship and other costs that are significantly in excess of those contemplated when the regulation was adopted. The proposed increase in the MSIV allowable leak rate will not be practical if the MSIV leak rate results are included in the Type A and Type C test acceptance criteria. Compliance requires unnecessary repair and retesting of the MSIVs. This significantly impacts the maintenance work load during plant outages and often contributes to outage extensions. The frequent disassembly and refurbishing of MSIVs, which is required to meet the low leakage limits, contributes to repeated failures.

Examples of these maintenance induced defects include machining-induced seat cracking, machining of guide ribs, excessive pilot valve seat machining, and mechanical defects induced by assembly and disassembly. By not having to disassemble the valves and refurbish them for minor leakage, BFN avoids introducing one of the root causes of recurring leakage. Industrial experience suggests that, by attempting to correct non-existing or minimal defects in the valves, it is likely that some actual defects may be introduced that lead to later leak test failures.

In addition, the frequent maintenance work results in needless dose exposures to maintenance personnel leading to additional economic burdens, and are inconsistent with As Low As Reasonably Achievable (ALARA) principles.

(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption.

Enclosure 1 contains an application for a license amendment which involves a proposed change to the Technical specifications to increase the allowable MSIVs leak rate. This application is partly based on the fact that the current limit is too restrictive, and results in excessive MSIV maintenance and repair, leading to additional MSIV failures, which in turn result in higher leakage. The proposed limit will benefit the public health and safety by reducing the potential for MSIV failures, and thus keeping the MSIV leakage within the radiological analysis values.

TVA proposes to implement the described reliable and effective method of utilizing the main steam piping and condenser for MSIV leakage treatment. This treatment method is effective to treat MSIV leakage over an expanded operating range without exceeding the off-site and control room dose limits. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any logic control and interlocks. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the environment. Therefore the proposed method is highly reliable for MSIV leakage treatment. The implementation will provide BFN with a capability to process MSIV leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage rate limit. From a safety perspective, the proposed changes result in an increase in protection to the public.

The exemption from Appendix J requirements for MSIV leakage rates is required so that BFN can operate with the proposed Technical Specifications increased MSIV allowable leakage values. This results in reduced radiological exposure to plant maintenance workers, greater overall MSIV reliability, and significant economic benefit to TVA and its customers as a result of reduced plant outage durations. These benefits will compensate for any decrease in safety that may result from the granting of the exemption.

Thus, as discussed above, special circumstances exist warranting the grant of the exemption.

VI. ENVIRONMENTAL IMPACT

The proposed exemption has been analyzed and determined not to cause additional construction or operational activities which may significantly affect the environment. It does not result in a significant increase in any adverse environmental impact previously evaluated, result in a significant change in effluents or power levels, or affect any matter not previously reviewed by the Nuclear Regulatory Commission which may have a significant adverse environmental impact.

The proposed exemption does not alter the land use for the plant, any water uses or impacts on water quality, air or ambient air quality. The proposed action does not affect the ecology of the site and vicinity and does not affect the noise emitted by station. Therefore, the proposed exemption does not affect the previous analysis of environmental impacts.

VII. CONCLUSION

Pursuant to 10 CFR 50.12, TVA requests an exemption from the requirements of 10 CFR 50, Appendix J, Option A, Sections II.H.4 and III.C.3, and 10 CFR 50, Appendix J, Option B. The exemption would allow the exclusion of the measured MSIV leakage from the combined local leak rate test results.

The BFN units were granted operating licenses pursuant to Section 104.b of the Atomic Energy Act and are not limited by statute to specific methods of testing primary containment integrity. Thus, the Commission can legally exempt TVA from the requirements of 10 CFR 50, Appendix J. The exemption does not present an undue risk to the public health and safety and is not inconsistent with the common defense and security. In addition, special circumstances are present which justify the exemption from this regulatory requirement. Specifically,

1. Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule.

2. Compliance would result in undue hardship and other costs that are significantly in excess of those contemplated when the regulation was adopted and that are significantly in excess of those incurred by others similarly situated.

3. There are also other material circumstance present that were not considered when the regulation as adopted for which it would be in the public interest to grant an exemption.

Therefore, in accordance with the provisions of 10 CFR 50.12, TVA requests an exemption to the requirements of 10 CFR 50, Appendix J, Option A, Sections II.H.4 and III.C.3, and 10 CFR 50, Appendix J, Option B, to allow the exclusion of the measured MSIV leakage from the combined local leak rate test results.

ENCLOSURE 5

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

EXEMPTION FROM 10 CFR 100, APPENDIX A, SECTION VI(a)

I. APPLICABLE RULE

The requirements of 10 CFR 100, Appendix A, Section VI(a) are as follows:

(1) Safe Shutdown Earthquake -

The nuclear power plant shall be designed so that, if the Safe Shutdown Earthquake occurs, certain structures, systems, and components will remain functional. These structures, systems, and components are those necessary to assure (I) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.... The engineering method used to insure that the required safety functions are maintained during and after the vibratory ground motion associated with the Safe Shutdown Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.

(2) Operating Basis Earthquake -

All structures, systems, and components of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public shall be designed to remain functional and within applicable stress and deformation limits when subjected to the effects of the vibratory motion of the Operating Basis Earthquake in combination with normal operating loads. The engineering method used to insure that these structures, systems, and components are capable of withstanding the effects of the Operating Basis Earthquake shall involve the use of either a suitable dynamic analysis or a suitable qualification test to demonstrate that the structures, systems and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate conservatism.

II. REQUESTED EXEMPTION

TVA requests an exemption from the requirements of 10 CFR 100, Appendix A, Section VI(a). The exemption would allow TVA to use alternate methods for the seismic evaluation of the capability of the main steam piping and condensers to process Main Steam Isolation Valve (MSIV) leakage following a design basis event coincident with a seismic event.

III. BACKGROUND

Pursuant to 10 CFR 50.12, TVA requests an exemption from the requirements of 10 CFR 100, Appendix A, Section VI(a). The exemption would allow TVA to use alternate methods for the seismic evaluation of the capability of the main steam piping and condensers to process MSIV leakage following a design basis event coincident with a seismic event. Specifically, TVA proposes to employ probability analysis, existing design capabilities, seismic experience, and a plant specific seismic adequacy verification as alternate methodology to the dynamic analysis or qualification test specified in Paragraph VI(a) of 10 CFR 100 Appendix A and to provide reasonable assurance that the existing main steam piping and condenser will remain functional following a design basis accident coincident with a significant seismic event. The exemption would allow the existing, non-seismically designed main steam piping and condenser to be used for mitigating the radiological consequences of MSIV leakage for the duration of a Design Basis Accident, such that the resulting doses are within the guidelines of 10 CFR 100 and General Design Criterion 19.

TVA proposes to utilize this reliable and effective method of utilizing the main steam piping and condenser for MSIV leakage treatment. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any logic control and interlocks. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the environment. Therefore, the proposed method is highly reliable for MSIV leakage treatment. The implementation will provide BFN with a capability to process MSIV leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage rate limit.

In conjunction with this application for an exemption request, Enclosure 1 contains an application for a license amendment to permit an increase in the allowable leak rate for the MSIVs. The safety analysis addresses the radiological effects of MSIV leakage following a postulated design basis loss of coolant accident (LOCA). TVA has demonstrated that the proposed change does not involve a significant hazards consideration. Based on this evaluation, TVA has determined the exemption to the requirements of 10 CFR 100, Appendix A, Section VI(a) is prudent and justified.

IV. TECHNICAL EVALUATION

The function of the primary containment is to isolate and contain fission products released from the reactor primary system following a design basis accident and to confine the postulated release of radioactive material. The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate. Periodic testing of the leak tightness of the primary containment as well as individual penetrations and valves is necessary to assure that the assumed release rate in the plants' safety analysis is conservative.

10 CFR 100, Appendix A, requires that structures, systems and components, which assure the capability to mitigate the consequences of accidents which could result in potential off-site exposures in excess of 10 CFR 100 limits, be designed to remain functional following a safe shutdown and operating basis earthquake (SSE and OBE) and concurrent loads. The BWROG has evaluated the capability of main steam piping and condensers to process MSIV leakage following a design basis accident coincident with a seismic event. Based on this comprehensive evaluation, the BWROG has concluded there is reasonable assurance that the main steam piping and condenser will remain functional following a design basis accident coincident with a design basis earthquake, to mitigate the radiological consequences of MSIV leakage. The following conclusions provide the bases for this assurance:

(1) The probability for which the resulting dose from MSIV leakage is significant is extremely low. This requires a design basis LOCA, a degraded core where the emergency core cooling systems are not functional, and a significant seismic event.

(2) The main steam piping and condensers are designed to strict industrial standards and building codes; thus, significant design margin exists.

(3) The main steam piping and condensers exhibit substantial seismic ruggedness. Comparisons of piping and condenser design in General Electric (GE) plants with those in the earthquake experience database reveal that the GE plant designs are similar to or more rugged than those that have exhibited good earthquake performance.

(4) The possibility of significant failure in GE BWR main steam piping or condensers in the event of design basis earthquakes is highly unlikely, and any such

failure would also be contrary to a large body of historical earthquake experience data, and thus unprecedented.

(5) A plant-specific verification of seismic adequacy of the main steam piping will be performed prior to their operation utilizing this exemption to provide reasonable assurance of the structural integrity of these components.

In support of the above, the BWROG has reviewed the potential combinations of LOCAs and seismic events of interest:

(1) LOCA WITHOUT NEAR COINCIDENT SEISMIC EVENT

For this occurrence the pressure in the piping system downstream of the MSIVs is rapidly reduced to atmospheric pressure; and since there is no seismic event, the flow path through main steam system piping to the condenser is assured.

(2) SEISMIC EVENT WITHOUT NEAR COINCIDENT LOCA

Without a LOCA and the potential associated core degradation, the radioactivity transported with MSIV leakage is of no radiological significance.

(3) LOCA WITH NEAR COINCIDENT SEISMIC EVENT

The consequences of this occurrence (also assuming significant core damage) are of interest because a seismic induced failure in the main steam or condenser system could allow MSIV leakage to bypass the treatment pathway. It has been previously well documented that the probability of a near coincident LOCA and seismic event is extremely small (design basis earthquake probability approximately 10^{-3} per reactor per year; the core damage frequency for Unit 2 with all three BFN units in operation is 2.8×10^{-5}, which is approximately a factor of 3.7 over the current single unit operation BFN Probabilistic Risk Assessment estimate of 7.6×10^{-6}). It is also noted that a LOCA does not induce a seismic event, and that a seismic event has a very low probability of causing a LOCA because the primary pressure boundary and emergency core cooling systems are designed to seismic requirements (NUREG/CR 4792 Volume 4 reported probability of seismic induced LOCA to be less than 5×10^{-7} per reactor per year).

Considering that the probability of a near coincident LOCA and seismic event is much smaller than other plant-safety risks (less than 1×10^{-7} per reactor per year for coincident events, less than 5×10^{-7} per reactor per year for seismic induced LOCA), the likelihood for main steam piping or condenser damage is extremely small.

Nevertheless, because main steam piping and condenser systems designs are extremely rugged, this equipment is expected to remain intact following design basis seismic events.

To further justify the capability of the main steam system piping and condenser treatment pathway, the BWROG has reviewed limited earthquake experience data on the performance of non-seismically designed piping and condensers (in past earthquakes). The study, documented in NEDC-31858, summarizes data on the performance of main steam piping and condensers in past strong-motion earthquakes and compares these piping and condensers with those in typical U.S. GE Mark I, II, and III nuclear plants. This limited earthquake experience data and similarity comparisons are then used to further strengthen the conclusions on how the GE piping and condensers would maintain their pressure retention function in a design basis earthquake in conjunction with a LOCA occurring just prior to or after the seismic event.

The earthquake experience data are derived from an extensive database on the performance of power plants and industrial facilities, compiled by EQE for the Seismic Qualification Utility Group, the Electric Power Research Institute, and many other EQE clients. This study summarizes the performance of over 100 power plant units (turbines, associated condensers, and main steam piping) in 19 earthquakes around the world from 1934 to the present.

The piping and condensers in the earthquake experience database exhibited substantial seismic ruggedness, even when they are not designed to resist earthquakes. This is a common conclusion in studies of this type on other plant items such as welded steel piping, anchored equipment such as motor control centers, pumps, valves, structures, and so forth. That is, with limited exceptions, normal industrial construction and equipment typically have substantial inherent seismic ruggedness, even when they are not designed for earthquakes. No failures of main steam piping were found. Anchored condensers have also performed well in past earthquakes with damage limited to minor internal tube leakage.

Comparisons of piping and condenser design in example GE Mark I, II, and III plants with those in the earthquake experience database reveal the GE plant designs are similar to or more rugged than those that exhibited good earthquake performance. The BWROG concludes that: (1) the possibility of significant failure in GE BWR main steam piping or condensers in the event of an eastern U.S. design basis earthquake is highly unlikely; and that (2) any such failure would also be contrary to a large body of historical earthquake experience data, and thus unprecedented.

Earthquake experience methodology has been applied in seismic equipment qualification issues associated with

Unresolved Safety Issue A-46 (Seismic Qualification of Equipment in Operating Plants). Piping performance data are presented in NUREG-1061 (a report from the NRC Piping Review Committee), and this report proposes changes to criteria that are directed toward the recognition of the superior performance of piping in earthquakes and establishes more realistic seismic criteria for piping qualification. The NRC has published NUREG-1030 and NUREG-1211 "Seismic Qualification of Equipment in Operating Nuclear Power Plants," which conclude that the seismic experience data approach provides the most reasonable and preferred alternative to other current equipment qualification methods.

The rapidly growing use of the seismic experience data approach is further illustrated by the fact that this method of analysis is now referenced in:

A. Draft Regulatory Guide 1.100, Revision 2, "Seismic Qualification of Electrical and Mechanical Equipment in Nuclear Power Plants,"

B. Recent approved revision of IEEE Standard 344-1987, "Recommended Practice for Seismic Qualification of Class 1E Equipment For Nuclear Power Generating Stations," and

C. Draft report of ASME Standard "Recommended Practice for Seismic Performance Qualification of Mechanical Equipment Used in Nuclear Power Plants."

The Seismic Qualification Utilities Group (SQUG) earthquake experience database includes a large number and variety of piping systems. In fact, piping is probably the strongest area in this regard (compared to areas like electrical or mechanical equipment, cable trays, etc.). It has been concluded that the earthquake experience data on piping, and in particular data on main steam piping, are applicable to main steam piping in BWRs.

In both nuclear and conventional power plants, the condenser is designed to reduce the low-pressure turbine outlet pressure (thereby increasing turbine efficiency) and to condense the steam. The nuclear environment does not impose additional significant design considerations on the condenser. With the exception of hotwell size, a conventional plant and nuclear plant with similar performance parameters have similar condensers.

None of the condensers within the seismic experience database has seismic design criteria. However, in view of the performance of the condensers within the database, it is concluded that the condensers have an inherent seismic ruggedness and that the earthquake experience data on condensers are applicable to condensers in BWRs.

Another recent study to develop, by data collection and statistical analysis, updated estimates of pipe breaks in commercial U.S. nuclear power plants was completed in 1987. This study evaluates both LOCA sensitive systems and non-LOCA sensitive systems. For BWR non-LOCA sensitive systems, ten pipe failures have occurred over 313 years of operating experience. None of these failures occurred in the main steam piping. Based on the observed failure rates, this study estimated the failure rate for the main steam system piping to be 7×10^{-4} failure/year/BWR with an upper bound of 9.6×10^{-3} failures/year/BWR. These results are consistent with the conclusion from the SQUG databases and NUREG-1169: BWR main steam piping is designed to withstand severe plant transients such as turbine trips and is expected to remain intact following accidents as severe as a design basis LOCA. Thus, the non-seismically designed main steam piping and the main condenser can be used to mitigate the consequences of MSIV leakage.

A plant-specific verification of seismic adequacy of the main steam piping will be performed prior to their operation utilizing this exemption will provide reasonable assurance of the structural integrity of these components.

In conclusion, there is reasonable assurance that the existing, non-seismically designed main steam piping and condenser will remain functional following a design basis accident coincident with a design basis earthquake, to mitigate the radiological consequences of MSIV leakage.

As discussed in Enclosure 1, plant specific radiological analyses have been performed to asses the effects of the proposed increase in the allowable MSIV leakage rate in terms of control room, Technical Support Center, and offsite doses following the worst case design basis accident (a double ended guillotine recirculation line break induced LOCA). The radiological analyses used the revised accident source term for light-water nuclear power plants contained in NUREG-1465 and conservative assumptions for the release of the source term. The contribution from the increased allowable MSIV leakage rate to the control room operator and offsite dose has been calculated separately. The analyses demonstrated that the resulting doses were below the regulatory limits contained in 10 CFR 100, Reactor Site Criteria, and 10 CFR 50, Appendix A, General Design Criterion 19, Control Room. Therefore, the exemptions to the requirements of 10 CFR 50, Appendix J, Sections II.H.4 and III.C.3 are acceptable and do not compromise the safety design basis of the primary containment or the overall purpose of performing leak rate testing.

Granting this exemption will not endanger life or property or the common defense and security. Granting this exemption would be in the public interest since it represents a significant cost reduction and would remove an unnecessary burden.

V. JUSTIFICATION FOR EXEMPTION

10 CFR 50.12 authorizes the NRC to grant exemptions from its own requirements. An exemption must (1) be authorized by law, (2) not present an undue risk to the public health and safety, (3) be consistent with the common defense and security, and (4) must entail special circumstances.

1. AUTHORIZED BY LAW

TVA was issued its operating licenses for BFN under the provisions of Section 104.b of the Atomic Energy Act. Operating licenses issued by the Commission pursuant to Section 104.b are not limited by statute to specific methods of the seismic evaluation of systems and components for adequacy. Thus, the Commission can legally exempt TVA from the requirements of 10 CFR 100, Appendix A, Section VI(a).

2. NOT PRESENT AN UNDUE RISK TO THE PUBLIC HEALTH AND SAFETY

The BWROG has evaluated the capability of main steam piping and condensers to process MSIV leakage following a design basis accident coincident with a seismic event. Based on this comprehensive evaluation, the BWROG has concluded there is reasonable assurance that the main steam piping and condenser will remain functional following a design basis accident coincident with a design basis earthquake, to mitigate the radiological consequences of MSIV leakage. This assurance is based on methodology using probability analysis, margins in the existing design codes, seismic experience, and a plant specific verification of seismic adequacy.

The treatment method for MSIV leakages is recommended by the BWROG in support of the resolution to Generic Issue C-8, "MSIV Leakage and LCS Failure." TVA proposes to implement the reliable and effective method for utilizing main steam piping and condenser for MSIV leakage treatment. This treatment method is effective to treat MSIV leakage over an expanded operating range without exceeding the off-site and control room dose limits. Except for the requirement to establish a proper flow path from the MSIVs to the condenser, the proposed method is passive and does not require any logic control and interlocks. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the environment. Therefore, the proposed method is highly reliable for MSIV leakage treatment. The implementation will provide TVA with a capability to process MSIV leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage rate limit. From a safety perspective, the proposed changes result in an increase in protection to the public. Therefore, the proposed exemption presents no undue risk to public health and safety.

3. BE CONSISTENT WITH THE COMMON DEFENSE AND SECURITY

The Commission's Statement of Considerations in support of the exemption rule note with approval the explanation of this standard as set forth in Long Island Lighting Company (Shoreham Nuclear Power Station, Unit 1), LBP-84-45, 20 NRC 1343, 1400 (October 29, 1984). There, the term "common defense and security" refers principally to the safeguarding of special nuclear material, the absence of foreign control over the applicant, the protection of Restricted Data, and the availability of special nuclear material for defense needs. The granting of the requested exemption will not affect any of these matters and, thus, such grants are consistent with the common defense and security.

4. MUST ENTAIL SPECIAL CIRCUMSTANCES

According to NRC regulations, special circumstances are present if any one of the six different cases cited in 10 CFR 50.12(a)(2) are present. TVA submits that the existence of special circumstances (ii), (iii) and (vi) are applicable for this exemption request:

(ii) Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

Strict compliance with Appendix A of 10 CFR 100 for the downstream main steam piping and condenser is not necessary to achieve the underlying purpose of the rule. The underlying purpose of the rule is to limit releases to within the off-site dose limits of 10 CFR 100. The regulation requires components that mitigate the consequences of an accident to within the dose limits of 10 CFR 100 be designed to the seismic requirements of 10 CFR 100, Appendix A. The regulation is intended to provide a reasonable assurance that the components will remain functional for the mitigating function. For the purpose of mitigating the radiological consequences of MSIV leakage, it is not necessary to apply the seismic requirements of 10 CFR 100, Appendix A to the main steam piping and condenser in order to achieve the underlying purpose of the rule because:

(1) There is reasonable assurance that the existing, non-seismically designed main steam piping and condenser will remain functional

following a design basis accident coincident with a design basis earthquake, to mitigate the radiological consequences of MSIV leakage. This assurance is based on methodology using probability analysis, margins in the existing design codes, seismic experience, and a plant specific verification of seismic adequacy.

(2) The safety analysis has been revised to assess the radiological consequences of MSIV leakage following a design basis LOCA. The analysis has demonstrated that the revised doses are well within the off-site dose guidelines of 10 CFR 100. Furthermore, the seismic approach is consistent with the current resolution of the seismic and equipment qualification issues. Earthquake experiences data have been applied in seismic equipment qualification issues associated with Unresolved Safety Issues A-46 (Seismic Qualification of Equipment in Operating plants). Piping performance data have been presented in NUREG-1061, a report from the NRC Piping Review Committee, which proposes changes to criteria that are directed toward the recognition of the superior performance of piping in earthquakes and establishes more realistic seismic criteria for piping qualification. The NRC has published NUREGs 1030 and 1211 "Seismic Qualification of Equipment in Operating Nuclear Power Plants," which conclude that the seismic experience data approach provides the most reasonable and preferred alternative to other current equipment qualification methods.

(iii) Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted, or that are significantly in excess of those incurred by others similarly situated.

The proposed MSIV leakage treatment method utilizes the existing main steam piping and condenser for the mitigating function. Compliance with the seismic requirements of 10 CFR 100, Appendix A for the main steam piping and condenser would require significant upgrade of the existing equipment, lead to unnecessary plant shutdown for modification, and significantly increase maintenance requirements and the associated costs in order to meet seismic qualification requirements.

(iv) The exemption would result in benefit to the public health and safety that compensates for any decrease in safety that may result from the grant of the exemption.

Enclosure 1 contains an application for a license amendment which involves a proposed change to the Technical specifications to increase the allowable MSIVs leak rate. This application is partly based on the fact that the current limit is too restrictive, and results in excessive MSIV maintenance and repair, leading to additional MSIV failures, which in turn result in higher leakage. The proposed limit will benefit the public health and safety by reducing the potential for MSIV failures, and thus keeping the MSIV leakage within the radiological analysis values.

TVA proposes to implement the described reliable and effective method for utilizing the main steam piping and condenser for MSIV leakage treatment. This treatment method is effective to treat MSIV leakage over an expanded operating range without exceeding the off-site and control room dose limits. Except for the requirement to establish a proper flow path from the MSIV to the condenser, the proposed method is passive and does not require any logic control and interlocks. The method is consistent with the philosophy of protection by multiple leak-tight barriers used in containment design for limiting fission product release to the environment. Therefore the proposed method is highly reliable for MSIV leakage treatment. The implementation will provide BFN with a capability to process MSIV leakage, and will also provide a uniform basis for establishing a plant-specific MSIV leakage rate limit. From a safety perspective, the proposed changes result in an increase in protection to the public.

The exemption from Appendix A requirements for the seismic evaluation of the capability of the main steam piping and condensers to process MSIV leakage following a design basis event coincident with a seismic event is required so that BFN can operate with the proposed Technical Specifications increased MSIV allowable leakage values. This benefit will compensate for any decrease in safety that may result from the granting of the exemption.

Thus, as discussed above, special circumstances exist warranting the grant of the exemption.

VI. ENVIRONMENTAL IMPACT

The proposed exemption has been analyzed and determined not to cause additional construction or operational activities which may significantly affect the environment. It does not result in a significant increase in any adverse environmental impact previously evaluated, result in a significant change in effluents or power levels, or affect any matter not previously reviewed by the Nuclear Regulatory Commission which may have a significant adverse environmental impact.

The proposed exemption does not alter the land use for the plant, any water uses or impacts on water quality, air or ambient air quality. The proposed action does not affect the ecology of the site and vicinity and does not affect the noise emitted by station. Therefore, the proposed exemption does not affect the previous analysis of environmental impacts.

VII. CONCLUSION

Pursuant to 10 CFR 50.12, TVA requests an exemption from the requirements of 10 CFR 100, Appendix A, Section VI(a). The exemption would allow TVA to use alternate methods for the seismic evaluation of the capability of the main steam piping and condensers to process MSIV leakage following a design basis event coincident with a seismic event.

The BFN units were granted operating licenses pursuant to Section 104.b of the Atomic Energy Act and are not limited by statute to specific methods of the seismic evaluation of systems and components for adequacy. Thus, the Commission can legally exempt TVA from the requirements of 10 CFR 100, Appendix A, Section VI(a). The exemption does not present an undue risk to the public health and safety and is not inconsistent with the common defense and security. In addition, special circumstances are present which justify the exemption from this regulatory requirement. Specifically,

1. Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule.

2. Compliance would result in undue hardship and other costs that are significantly in excess of those contemplated when the regulation was adopted and that are significantly in excess of those incurred by others similarly situated.

3. There are also other material circumstance present that were not considered when the regulation as adopted for which it would be in the public interest to grant an exemption.

Therefore, in accordance with the provisions of 10 CFR 50.12, TVA requests an exemption to the requirements of 10 CFR 100, Appendix A, Section VI(a) in order to allow TVA to use alternate methods for the seismic evaluation of the capability of the main steam piping and condensers to process MSIV leakage following a design basis event coincident with a seismic event.

ENCLOSURE 6

TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT (BFN)
UNITS 1, 2 AND 3

SUMMARY OF COMMITMENTS

1. Details regarding the methods for controlling suppression pool pH after a design basis accident will be provided.

2. The results of the evaluation for the seismic adequacy of the turbine building and the main steam piping and components downstream of the main steam isolation valves will be submitted for each unit.

LICENSING TRANSMITTAL TO NRC
SUMMARY AND CONCURRENCE SHEET

A concurrence signature reflects that the signatory has assured that the submittal is appropriate and consistent with TVA Policy, applicable commitments are approved for implementation, and supporting documentation for submittal completeness and accuracy has been prepared.

DATE 12/14/95 DATE DUE NRC 12/22/95 "A"

SUBMITTAL PREPARED BY (1) S. M. Kane ____________
Name Signature

SUBJECT Units 1, 2 and 3 Technical Specification Change TS 356, Increase in Allowable MSIV Leakage Rate and Request for Exemptions

Does this submittal contain Corrective Action/Commitment? X Yes ___ No

INDEPENDENT REVIEW (2) DATE ____________

CONCURRENCE (3)

NAME	ORGANIZATION	SIGNATURE	DATE
	RARC Chairman		
	PORC Chairman		
J. M. Corey	RadChem		
J. E. Maddox	Maintenance		
J. E. McCarthy	Mech./Nuc. Eng.		
T. J. McGrath	NSRB Chairman		
R. J. Moll	Operations		
G. D. Pierce	Tech. Support		
E. Preston	Plant Manager		
Pedro Salas	Site Licensing		
T. D. Shriver	NA&L Manager		
J. Valente	Civil Eng.		
E. J. Vigluicci	OGC		
H. L. Williams	Eng. & Matls. Mgr.		