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FROM: Chet Poslusny

SUBJECT: LOCA ANALYSIS

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Subject LOCA

15.6.5.5 Radiological Consequence

Two specific analyses are provided for the evaluation of the radiological consequences of a design basis Loss of Coolant Accident (LOCA), one for offsite dose evaluations and the second for control room dose evaluations. Both analyses are based upon assumptions provided in Regulatory Guide 1.3 except where noted. The analysis is based upon a process flow diagram shown in Figure 15.6-2 and accident parameters specified in Table 15.6-8.

15.6.5.5.1 Fission Product Release and Pathways

Fission product releases are based upon Regulatory Guide 1.3 in that it is assumed that of the fission products found in the core, 100% of the noble gases and 50% of the iodines are released from the core. Of these iodines, 50% are assumed to plate out leaving 25% of the total core inventory of iodine airborne and available for release. The chemical species differentiation for the iodine isotopes released to the containment atmosphere are assumed as specified in Regulatory Guide 1.3 as 91% elemental form, 4% organic form, and the remaining 5% as particulate form. Following the release of fission products to the containment atmosphere from the reactor pressure vessel, the fission products are subject to hold-up and radioactive decay, removal processes, and leakage to other plant areas and to the environment.

Two specific pathways are analyzed in releasing fission products to the environment. The first pathway is leakage to the reactor building (secondary containment) via penetrations and engineered safety feature components. This leakage pathway is assumed as not greater than an equivalent release of 0.5% by volume per day of the primary containment free air volume per plant technical specification. The secondary containment is a multi-compartment self contained structure maintained at negative pressure with respect to the environment thereby providing a significant hold up volume for fission product releases.. All leakage pathways from the primary containment except the main steamlines and the feedwater lines terminate in the reactor building. Leakage through the steamlines is treated separately below and leakage through the feedwater lines is assumed negligible assuming the proper isolation and filling of the feedwater lines upstream of the primary containment through the feedwater system. Flow through the reactor building/secondary containment is directed via the standby gas treatment system to the plant stack through hepa and charcoal filters. Credit is taken for hold up assuming 50% mixing in the secondary containment without plateout and other removal processes except filtration in the stand by gas treatment system (SGTS) as given in Table 15.6-8. It is assumed that for the first 20 minutes after an isolation signal the SGTS is drawing the reactor building down to negative pressures, and therefore all leakage during this time period is assumed without effective filtration. Following this 20 minute period, full filtration is assumed for the remainder of the period.

Removal process in the primary containment and for leakages from the primary containment are described in the following sections. Section 15.6.5.5.1.1 discusses reductions is airborne iodine due to water attrition while sections 15.6.5.5.1.2 and 15.6.5.5.1.3 discuss removal processes for leakages downstream of the main steamline isolation valves.

15.6.5.5.1.1 Suppression Pool Scrubbing

The BWR suppression pool, though designed primarily as a pressure suppression mechanism for vessel blow down, serves also as an excellent medium for the intrainment and capturing of all fission products except the noble gases. The design and operational characteristics of the BWR provide for a release pathway from the vessel and drywell into the suppression pool for all cases involving vessel depressurization and therefore for removal of fission products by scrubbing in the suppression pool. The NRC has accepted the fact that the suppression pool is capable of removing fission products and provides for credit to incorporate this phenomena in design basis analysis by recourse to the requirements of Standard Review Plan 6.5.5. The requirements of SRP 6.5.5 state that any flow directed through the pool can be credited with a decontamination factor of 10 providing the requirements of subsection II are met and that the total decontamination is a combination of the decontamination applied to flow through the pool to that fraction of the release which bypasses the pool. The following paragraphs describe the determination of the bypass fraction for the calculation of overall pool decontamination.

The requirements of Regulatory Guide 1.3 stipulate an instantaneous release of fission products from the vessel to the containment atmosphere. Coincident with an instantaneous release, under LOCA conditions, the BWR pressure vessel will be depressurized resulting in the purging of the primary containment atmosphere to the suppression pool. This situation is shown in Figure 15.6-3 which show the fractions of airborne particulate as a function of time in the drywell and wetwell airspaces assuming a decontamination factor of 10 for that flow which is purged either through the horizontal vents or the safety relief valves. The figure shows that the airborne inventory is reduced by almost a factor of ten within two minutes of the initiation of the blowdown event.

However, the application of the precepts of Regulatory Guide 1.3 do not indicate the most likely train of events in a core damage event which is what is implied in the design basis release assumptions. Both Regulatory Guide 1.3 and its predecessor, TID-14844, are based upon non-mechanistic assumptions and devices and are in the process of being replaced. Therefore consideration of a range of accident progressions beyond the rigidly narrow scope of Regulatory Guide 1.3 is given below to evaluate potential suppression pool bypass under more realistic conditions.

The basic assumptions of this evaluation of suppression pool bypass conditions assumes that an event occurs which challenges the reactor core causing sufficient damage to release approximately half the fission product volatile iodines. Damage to the core is limited to this extent implying the ability to recover core cooling and limit in-vessel damage. Such an assumption complies with the intent of design basis licensing in that the exact means by which the core is challenged is not specified but given the challenge, the response and adequacy of the plant design is tested. In addition, the assumption of resumpiton of core cooling and recovery with limited release is fully justifiable since the ABWR incorporates multiple cooling modes with redundant safety grade cooling systems. Events leading to more significant core damage are not considered as design basis since they assume massive damage with multiple failures to the design safety systems. Such events are of exceedingly low probability and are described and evaluated in Chapter 19. Therefore broadly speaking events which lead to the assumed damage can be divided into two categories, break and non-break. Break events are those through which primary coolant are released directly to the primary containment atmosphere and non-break events are those in which the primary coolant boundary is not breached. Both types of events will be considered below to provide a bounding analysis for suppression pool bypass.

In considering the non-break events, core damage is primarily the result of failure to maintain proper core water level resulting in uncovering the core with subsequent release of fission products upon overheating of the fuel rods. To consider the train of events in such a case, the MAAP code (see subsection 19.E for a description of the MAAP code) was used to model vessel response. Based upon the MAAP analysis, releases would begin shortly after core water level reaches the bottom of the core and would proceed rapidly. During this period isolation of the primary coolant system and containment would have been automatically tripped on low water level and the main steam isolation valves as well as all the other isolation valves would have tripped effectively isolating all flow from the primary containment. Therefore, the released fission products would be exposed to three primary influences: (1) plateout and removal in the dryers and separators, (2) leakage from the main steamline isolation valves into the main steamlines, and (3) flow through the safety relief valves into the suppression pool.

The release of volatile fission products would occur over a period of 10-20 minutes during which steam or hydrogem flow from the core region would be very small. Using an upper bound estimate of 2 kg/sec of steam generation during this period, the vessel flushing rate would be once every ten minutes. Therefore during this period 0.13% of the flow would bypass the pool through MSIV leakage. The remaining faction would be transported through the safety relief valves. Without recovery of cooling water after this period significant damage would occur to the core beyond that of a design basis event. With the recovery of water, the energy generated from decay heat which would be evident in overall core temperature rise and core degradation would cause a rapid pulse of steam resulting in the purging the pressure vessel of all airborne materials. Based upon the MAAP analysis it is conservatively estimated that 9×10^3 Kg of steam would be generated in a short period of time on the order of minutes resulting in a vessel purge rate of seven to eight complete exchanges. Therefore effectively all fission products remaining airborne in the vessel or lines would be purged to the suppression pool. The effective pool bypass fraction would then be 0.13% for an integrated overall DF of 9.8 without credit for plateout or 4.9 with a factor of two plateout.

The break case follows a similar logic. Initially, following a break massive depressurization of the pressure vessel would occur causing all non-condensables in the drywell to be purged into the wetwell air space through both the horizontal vents and the safety relief valves. Isolation of the containment and associated lines would be automatically initiated on depressurization. Following this rapid depressurization there would follow a period during which the water level in the vessel would drop to the bottom of core resulting in the eventual release of fission products form the core. Since in a break case, the path of least resistance would be through the break, the fission products would be effectively purged to the drywell airspace. In this case the temperatures and surface areas involved would provide adequate plate out areas to validate the Regulatory Guide 1.3 plate out factor of 2. Like the non-break case the total release is limited implying resumption of cooling and a massive release of steam upon resumption of cooling. In the case of reflood with a break, because of the large volume of the drywell, convervatively 80% of the drywell volume is purged during the reflood period. If complete mixing is assumed which is resonable because of the dynamic flows involved, it is then found that 55.6% of the airborne fission products are purged to the suppression pool in the few minutes needed to reflood the core. Therefore in this case an integrated pool DF of 2 is calculated.

In summary, it is found that for design basis accident conditions in which credit is taken for the proper operation of redundant safety grade systems subject to the single failure proof criteria that the suppression pool is capable of reducing the elemental and particulate airborne iodine inventory by a factor of 2.

15.6.5.5.1.2 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exits which permits the primary containment atmosphere or in the non-break case pressure vessel air space direct access to the main steamlines and that the main steamline isolation valves leak at the maximum technical specification. Furthermore, it is assumed that the most critical main steamline isolation valve fails in the open position. Therefore, the total leakage through the steamlines is equal to the maximum technical specification for the plant.

The main steamlines are graded (see Table 3.2-1) as Seismic Classification I Quality Group B from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV thereby providing a qualified safety grade mitigation system for fission product leakage which in this case is limited by the leakage criteria specific in the technical specifications for the Main Steamline Isolation Valves (MSIV). The primary purpose of this system is to stop any potential flow through the main steamlines. Down stream of the seismic restraint referred to above, the steamlines pass through the reactor building - control building interface into the steam tunnel located in the control building upper floor. This steam tunnel is a heavily shielded seismic category I structure designed primarily to shield the control building complex. From the control building the steamlines pass through the control building - turbine building interface into the turbine building steam tunnel which is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines, their associated branch lines outboard of the last reactor building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Table 3.2-2, note R) which determines the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the primary containment through the main steamlines involves the determination of (1) probable and alternate flow pathways, (2) physical conditions in the pathways, and (3) physical phenomena which affect the flow and concentration of fission products in the pathways. The most probable pathway for fission product transport from the main steamlines is found to be from the outboard MSIV's into the drain lines coming off the outboard MSIV and then into the turbine building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines. Consideration of the main steamlines and drain line complex downstream of the reactor building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination.

1. The main steamlines and drain lines are high quality lines inspected on a regular schedule.

2. The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.

3. The main steamlines and drain lines are enclosed in a shielded corridor which protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as built inspection is required to verify that no damage could be expected from other components and structures in a SSE.

4. The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to assure their survival under SSE conditions. A large data base of experience in the survival of these types of components under actual earthquake conditions exits which prove this contention. (Reference 5) In the case of ABWR further margin for survival can be expected since the ABWR lines are designed through dynamic analysis to survive such events whereas in the case of the actual experience data base, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Therefore, based upon the facts above, the main steamlines and drain lines in the ABWR are used as mitigative components in the analysis of leakage from the MSIVs.

The analysis of leakage from the MSIVs follows the procedures and conditions specified in Reference 5. Two flow paths are analyzed for dose contributions. The first pathway through the drain lines is expected to dominated due to the incorporation of a safety grade isolation valve on the outboard drain line which will open the line for flow down the drain line under LOCA conditions. The second pathway through the main steamlines into the turbine is expected to carry less than 0.3% of the flow based upon a determination that the maximum leakage past the turbine stop valves with an open drain line would permit only 0.3% flow for the valves to operate within specification. Specific values used and results of the main steamline leakage analysis are given in Table 15.6-8.

15.6.5.5.1.3 Condenser and Turbine Modeling

The condenser and turbine are modeled as detailed in Reference 5 with specific values used given in Table 15.6-8. Both volumes are modeled primarily as stagnant volumes assuming the shutdown of all active components. Both turbine and condenser are used as mitigative volumes based upon the determination that such components designed to standard engineering practice are sufficiently strong to withstand SSE conditions due wholly to their design. (Reference 5) The only requirement in the design of the condenser being that it be bolted to the building basemat to prevent walking during an earthquake. The turbine has no such restriction and may possibly move. The requirement on these components for purposes of mitigation is only that they survive as a volume and not that they provide functionality or leak tightness following an earthquake.

Release from the condenser/turbine building pathway are assumed via diffuse sources in the turbine building. The two major points of release in the turbine building are expected to be the truck doors at the far end of the turbine building and the maintenance panels located midway on the turbine building on the side opposite the service building. Releases are assumed to be ground level releases. See section 15.6.5.5.3 for applicable meteorology.

15.6.5.5.2 Control Room

The ABWR control room is physically integrated with the reactor building and turbine buildings and is located between these structures (see Figure 15.6-4). During a LOCA, exposure to the operators will consist of contributions from airborne fission products entrained into the control room ventilation system and gamma shine from the reactor building and airborne fission products external to the control building. Of these contributions, the last two involving gamma shine are negligible since the inhabited portions of the ABWR control room are physically located underground with sufficient shielding overhead (a minimum of 1.6 meters of concrete) and in the side walls (1.2 meters) to protect the operators from the normal steamline gamma shine. Such shielding is more than sufficient to protect the operators given any amount of airborne fission products.

Therefore, exposure to the operators will consist almost entirely of fission products entrained into the control room environment from the atmosphere. The ABWR control room uses a redundant safety grade HVAC system with two inch charcoal filters for removal of iodines and two roof mounted automatically controlled intake vents. The location of the vents are given in Figure 15.6-4. Because of the location of these vents, it cannot be assumed that at least one vent will be uncontaminated given most conditions of meteorology. Therefore, complete credit for dual intakes was not taken and only a partial credit of a factor of four reduction in control room dose was assumed even though both intake vents are computer controlled for minimum radiation selection. In addition, the location of these vents with respect to the potential release points show that given any wind flow condition, the vents may be contaminated only by a release from the reactor building or turbine building but not both. Nevertheless, for purposes of conservative calculations, it was arbitrarily assumed that for 30% of the time stagnant meteorological conditions were assumed such that the primary intake vent was contaminated by both sources. For the remaining 70% of the time only the more significant source was assumed to contaminate the primary intake vent.

Infiltration of airborne contamination to the control room was considered negligible owing to the pathway for access to the control complex. Entry into the control room is via the service building and a labyrinth doorway entry system through double doors into the clean portions of the service building. From the service building additional controlled access through double doors provides entry into the control room. In each of these entry/access door systems, positive pressure is maintained to vent infiltrated air to the outside and away from the control room complex. As such no contamination is anticipated beyond the initial access entry way from which infiltrating air is purged to the environment.

Control room dose is based upon fission product releases modeled as described in paragraph 15.6.5.5.1 and the values presented in Table 15.6-8. Operator exposure was based upon those conditions given in Table 15.6-8 and occupancy factors as shown below derived from SRP 6.4. Meteorology was derived as is specified in section 15.6.5.3.2.

Time	Occupancy Factor
0-1 day	1.0
1-4 days	0.6
> 4 days	0.4

15.6.5.5.3 Meteorology and Site Assumptions

15.6.5.5.3.1 Offsite Meteorology

The SSAR involves the use of a generic U.S. site which does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore for the evaluation of offsite accident conditions recourse was made to Regulatory Guides 1.145 and 1.3 for meteorological definitions. Specifically, the table found in section C.2.g(4) of Regulatory Guide 1.3 was used to define the meteorological parameters for use with the models found in Regulatory Guide 1.145. All releases were defined as ground level incorporating building wake conditions using the minimum ABWR building cross section.

Unlike the other design basis accidents found in Chapter 15, the LOCA accident analysis requires the development of meteorological conditions over a 30 day period. Therefore the specification of a maximum value for the χ/Q dispersion parameter is not feasible for the 30 day analysis. Instead, for the determination of site suitability with respect to the offsite dose for LPZ calculations, a table of multiplicative values varying by time is provide under table 15.6-14. Site specific χ/Q's in sec/m^3 for the indicated time periods should be multiplied by the respective multiplier and summed to determine the LPZ 30-day dose.

15.6.5.5.3.2 Control Room Meteorology

No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. Unlike the offsite dose methodology which is a relatively straight forward application of Regulatory Guides 1.3 and 1.145, the parameters and methods by which the control room intake concentrations can be calculated are poorly characterized and currently not codified in a usable form. Therefore, for application to ABWR, a back calculation was used to provide an estimate of the meteorological χ/Q dispersion parameters which would provide for the maximum acceptable dose under SRP 6.4. Since the calculation covers a period of 30 days, a variation in meteorological χ/Q was assumed for variations in wind direction and wind speed. The variation factors chosen were take from Table 1 of Reference 4 and are shown below.

Time Period	Murphy-Campe χ/Q Improvement Factor
0-8 hours	1.0
8-24 hours	0.59
1-4 days	0.375
> 4 days	0.165

Also, since the control room may be contaiminated from two physically separated sources, the reactor building stack base or the turbine building truck doors, reference was made to the most recently published work of Ramsdell to evaluate the differences in χ/Q for releases from each source to the control building. Using the methodology given in Reference 7, it was determined that releases from the turbine building at 108 meters from the control room intake would be a factor of six lower in concentration for a equal release than releases from the reactor building stack base at 41 meters from the nearest control building intake. Therefore a factor of six improvement in χ/Q was assumed for releases from the turbine building.

For application to specific site analysis two methods exist for determination of control room dose. The first method is a one on one comparison of the χ/Q values in Table 15.6-14 to the site χ/Q's. If the site χ/Q's are for all values less than the values in Table 15.6-14, then the control room doses are less than regulatory requirements. If this is not true then a site specific calculation needs to be performed for the site. For this purpose, an isotope by isotope release rate table is given in Table 15.6-10 and 15.6-12 from which actual calculations can be made.

15.6.5.5.4 Results

The results of this analysis are presented in Tables 15.6-13 and 15.6-14 for both offsite and control room dose evaluations and are within current regulatory guidelines.

15.6.7 References

5. L.S. Lee, *BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems*, February, 1991, NEDC-31858P.

7. Ramsdell, J.V., *Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes*, 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

Table 15.6-8

LOSS OF COOLANT ACCIDENT PARAMETERS

I Data and Assumptions used to esimate source terms.

A.	Power Level	4005 MWt
B.	Fraction of Core Inventory Released	
	Noble Gases	100%
	Iodines	50%
C.	Iodine Initial Plateout Fraction	50%
D.	Iodine Chemical Species	
	Elemental	91%
	Particulate	5%
	Organic	4%
E.	Suppression Pool Decontamination Factor - sec 15.6.5.5.1.1	
	Noble Gas	1
	Organic Iodine	1
	Elemental Iodine	2
	Particulate	2
	Pool Bypass Area	0.05 ft^2

II Data and Assumption used to estimate activity released.

A.	Primary Containment Leakage	
	(1) Penetration and ESF Equipment	0.5%/day
	(2) MSIV Leakage (Total all lines)	140 SCFH
B.	Reactor Building Leakage	
	(1) 0-20 min	150%/day
	(2) >20 minutes	50%/day
C.	SGTS	
	Filter Effeciency (6 inch charcoal)	97%
	Drawdown Time	20 min
D.	MSIV Leakage - see Reference 5 for standard parameters	
	Main Steam Line Length	157 ft
	Drain Line Length	235 ft
	Main Steam Line IR/OR	31.98/35.55cm
	Drain Line IR/OR	3.33/4.45cm
	Main Steam Line Insulation	12.0 cm
	Drain Line Insulation	6.5cm
	Plateout and Resuspension Factors	Ref 5.
E.	Condenser Data	
	Free Air Volume	220,000ft^3
	Fraction of Volume involved	20%
	Leakage Rate	11.6%/day
	Iodine Removal Factor	
	Elemental	0.993
	Particulate	0.993
	Organic	0

Table 15.6-8 (cont'd)

III Control Room Data

A.	Control Room Volumes	
	Total Free Air Volume	7,000 m^3
	Gamma Room Volume (room size)	1,200 m^3
B.	Recirculation Rates	
	Filtered Intake	1.8 m^3/sec
	Unfiltered Intake	0.0
	Filtered Recirculation	0.83 m^3/sec
	Filter Effeciecy (2 inch charcoal)	95%

IV Dispersion and Dose Data

A.	Meteorology	Sec 15.6.5.5.3 Tbls 15.6-13,14
B.	Dose Calculation Method (semi-infinite)	Ref 2 & 3, RG 1.109
C.	Dose Conversion Assumptions	Ref 2, 3
D.	Activity/Releases	Tbls 15.6-9,10,11,12 Appendix A.
E.	Dose Evaluation	Tbls 15.6-13,14

Table 15.6-10 Iodine Activity Release to Environment

A. Activity Released from Reactor Building

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 Days
I-131	2.03E+02	2.06E+02	2.54E+02	6.12E+02	3.91E+03	1.69E+04
I-132	2.78E+02	2.81E+02	2.96E+02	3.03E+02	3.04E+02	3.04E+02
I-133	4.22E+02	4.27E+02	5.14E+02	9.68E+02	..26E+03	2.45E+03
I-134	3.94E+02	3.[illegible]6E+02	3.99E+02	3.99E+02	3.99E+02	3.99E+02
I-135	3.92E+02	3.97E+02	4.52E+02	5.91E+02	6.55E+02	6.55E+02
TOTAL	1.69E+03	1.71E+03	1.92E+03	2.87E+03	7.53E+03	2.07E+04

B. Activity Released from Condenser in Curies

B.1 Elemental and Particulate Releases

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 Days
I-131	2.00E-06	1.46E-05	3.65E-04	3.27E-03	1.65E-02	2.26E-02
I-132	2.16E-06	1.23E-05	1.01E-04	1.46E-04	1.47E-04	1.47E-04
I-133	4.05E-06	2.89E-05	6.36E-04	4.15E-03	1.03E-02	1.05E-02
I-134	2.09E-06	8.11E-06	2.24E-05	2.29E-05	2.29E-05	2.29E-05
I-135	3.56E-06	2.39E-05	3.92E-04	1.36E-03	1.76E-03	1.76E-03
Total	1.39E-05	8.78E-05	1.52E-03	8.95E-03	2.87E-02	3.50E-02

B.2 Organic Release

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 Days
I-131	2.41E-05	1.77E-04	4.42E-03	4.01E-02	2.43E-01	5.29E-01
I-132	2.61E-05	1.49E-04	1.22E-03	1.77E-03	1.78E-03	1.78E-03
I-133	4.90E-05	3.50E-04	7.71E-03	5.09E-02	1.37E-01	1.41E-01
I-134	2.53E-05	9.81E-05	2.70E-04	2.76E-04	2.76E-04	2.76E-04
I-135	4.31E-05	2.89E-04	4.76E-03	1.66E-02	2.18E-02	2.18E-02
Total	1.68E-04	1.06E-03	1.84E-02	1.10E-01	4.04E-01	6.94E-01

B.3 Organic Release from Resuspended Sources

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 Days
I-131	0	7.21E-04	3.86E-02	1.00E+00	1.70E+02	1.38E+04
I-132	0	5.60E-04	7.86E-03	1.85E-02	1.94E-02	1.94E-02
I-133	0	1.41E-03	8.48E-01	1.17E+00	3.84E+01	8.92E+01
I-134	0	3.43E-04	1.33E-03	1.52E-03	1.53E-03	1.53E-03
I-135	0	1.14E-03	3.70E-02	3.11E-01	9.27E-01	9.41E-01
Total	0	4.18E-03	9.33E-01	2.50E+00	2.10E+02	1.39E+04

Table 15.6-12 Noble Gas Released to Environment

A. Activity Released from Reactor Building in Curies

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 days
KR-83M	1.72E+02	1.73E+02	1.80E+02	1.82E+02	1.82E+02	1.82E+02
KR-85M	3.87E+02	3.91E+02	4.34E+02	5.01E+02	5.13E+02	5.13E+02
KR-85	1.79E+01	1.81E+01	2.25E+01	5.61E+01	4.21E+02	4.30E+03
KR-87	6.82E+02	6.86E+02	7.00E+02	7.01E+02	7.01E+02	7.01E+02
KR-88	1.03E+03	1.04E+03	1.11E+03	1.17E+03	1.17E+03	1.17E+03
KR-89	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02	1.40E+02
XE131M	9.40E+00	9.52E+00	1.18E+01	2.87E+01	1.93E+02	1.05E+03
XE133M	1.37E+02	1.38E+02	1.69E+02	3.74E+02	1.62E+03	2.71E+03
XE-133	3.28E+03	3.32E+03	4.10E+03	9.70E+03	5.71E+04	1.82E+05
XE135M	3.55E+02	3.55E+02	3.55E+02	3.55E+02	3.55E+02	3.55E+02
XE-135	4.18E+02	4.23E+02	4.91E+02	7.19E+02	9.12E 02	9.13E+02
XE-137	4.06E+02	4.06E+02	4.06E+02	4.06E+02	4.06E+02	4.06E+02
XE-138	1.48E+03	1.48E+03	1.49E+03	1.49E+03	1.49E+03	1.49E+03
TOTAL	8.52E+03	8.59E+03	9.61E+03	1.58E+04	6.52E+04	1.96E+05

B. Noble Gas Release from Condenser in Curies

Isotope	1 Hour	2 Hours	8 Hours	1 Day	4 Days	30 days
Kr-83M	1.99E-04	1.07E-03	7.12E-03	9.14E-03	9.16E-03	9.16E-03
Kr-85	2.78E-05	2.04E-04	5.18E-03	4.89E-02	3.40E-01	1.20E+00
Kr-85M	5.30E-04	3.40E-03	4.57E-02	1.13E-01	1.26E-01	1.26E-01
Kr-87	6.89E-04	3.22E-03	1.39E-02	1.52E-02	1.52E-02	1.52E-02
Kr-88	1.32E-03	7.83E-03	7.67E-02	1.29E-01	1.32E-01	1.32E-01
Kr-89	4.29E-09	4.29E-09	4.29E-09	4.29E-09	4.29E-09	4.29E-09
Xe-131M	1.45E-05	1.07E-04	2.68E-03	2.46E-02	1.56E-01	3.86E-01
Xe-133	5.06E-03	3.70E-02	9.18E-01	8.16E+00	4.63E+01	8.52E+01
Xe-133M	1.99E-04	1.07E-03	7.12E-03	9.14E-03	9.16E-03	9.16E-03
Xe-135	6.10E-04	4.20E-03	7.78E-02	3.44E-01	5.43E-01	5.43E-01
Xe-135M	6.78E-05	9.82E-05	1.02E-04	1.02E-04	1.02E-04	1.02E-04
Xe-137	8.24E-08	8.24E-08	8.24E-08	8.24E-08	8.24E-08	8.24E-08
Xe-138	2.27E-04	3.03E-04	3.10E-04	3.10E-04	3.10E-04	3.10E-04
Total	8.94E-03	5.85E-02	1.15E+00	8.85E+00	4.76E+01	8.76E+01

Table 15.6-13

LOSS OF COOLANT ACCIDENT

METEOROLOGY AND OFFSITE DOSE RESULTS

SITE BOUNDARY DOSE RESULTS

Meteorology* (sec/m^3)	Dist (m)	Thyroid Dose (Rem)	Whole Body Dose (Rem)
2.76E-03	max	300	5.4E+00
1.18E-03	300	1.3E+02	2.3E+00
2.19E-04	800	2.4E+01	4.3E-01

LOW POPULATION ZONE BOUNDARY DOSE RESULTS

Time (hrs)	Meteorology (sec/m^3)	Dist (m)	Thyroid Dose (Rem)	Whole Body Dose (Rem)
8	3.73E-05	4828	5.4E+00	8.8E-02
24	1.21E-05		7.2E+00	1.1E-01
96	4.27E-06		1.6E+01	1.5E-01
720	9.09E-07		3.0E+01	2.8E-01

* "max" = maximum meteorology to meet 10CFR100 limitation.

Table 15.6-14

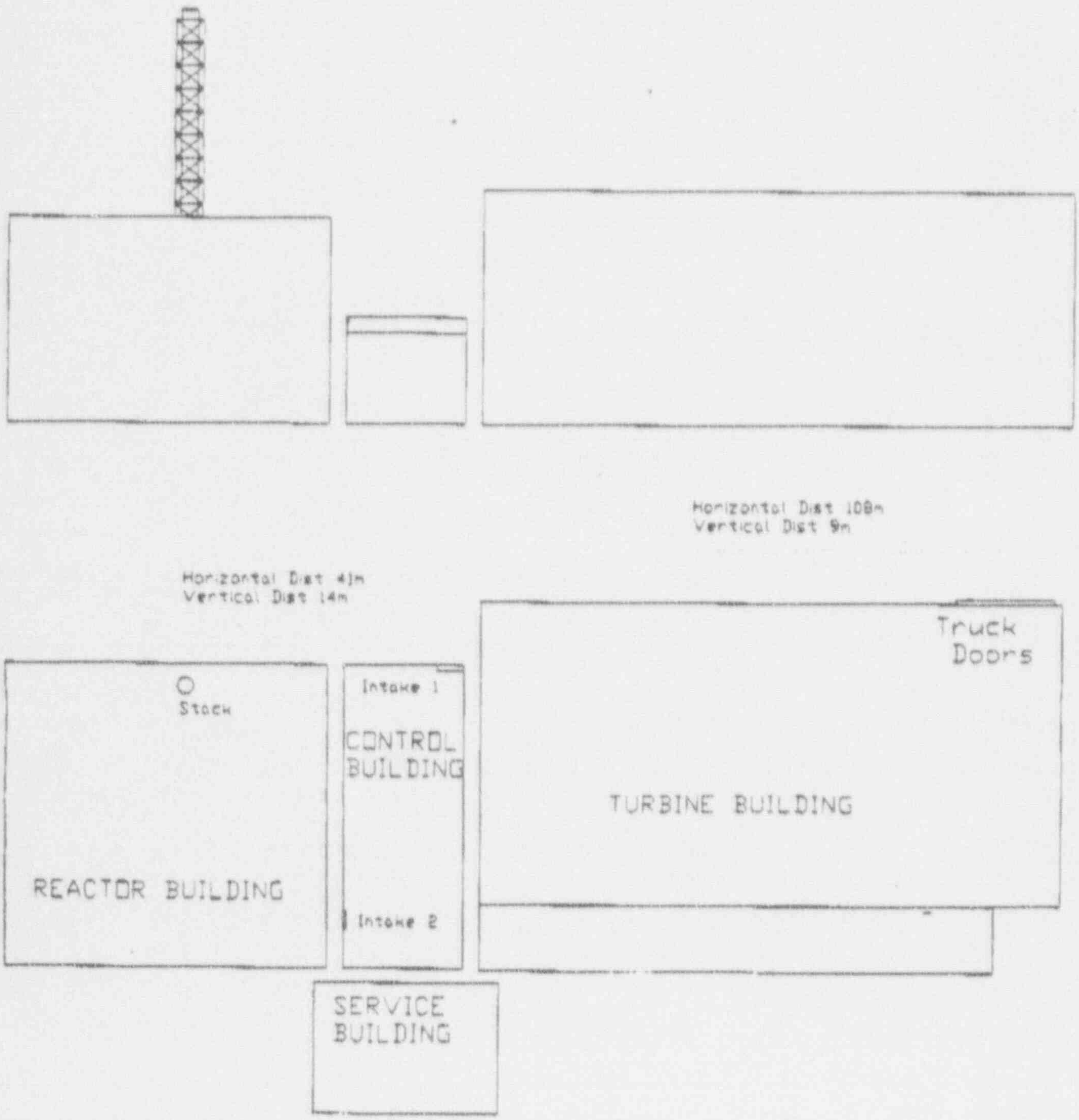
LOSS OF COOLANT ACCIDENT

METEOROLOGY AND CONTROL ROOM DOSE RESULTS

Time	Meteorology* (sec/m^3)	Thyroid (Rem)	Whole Body (Rem)	Beta (Rem)
0-8 hrs	4.00E-03	4.1E-01	3.9E-01	8.4E-01
8-24hrs	2.36E-03	7.2E+00	9.6E-01	2.5E+00
1-4days	1.50E-03	1.7E+01	2.4E+00	7.0E+00
4-30days	6.60E-04	29.5	3.5E+00	1.1E+01

*See Subsection 15.6.5.5.3.2 for description of meteorology. Values are for dispersion from Reactor Building. Dispersion values for releases from Turbine Building are a factor of six less than Reactor Building dispersion values.

FIGURE 15.6-4 ABWR PLANT LAYOUT



APPENDIX A

TO

SECTION 15.6.5

THIS APPENDIX PROVIDES ADDITIONAL DETAIL ON THE DISTRIBUTION OF IODINE ISOTOPES FOR THE DESIGN BASIS LOCA ANALYSIS FOUND IN SUBSECTION 15.6.5. THE INFORMATION IS IN THE FORM OF A SERIES OF GRAPHS AS IS EXPLAINED BELOW.

GRAPH	EXPLANATION
A-1	Provides the total airborne fraction of iodine in the primary containment as a function of time.
A-2	Provides the total airborne fraction of iodine in the reactor building as a function of time.
A-3	Provides the distribution of elemental (including elemental and particulate) and organic iodine in the condenser which originated in the primary containment as a function of time.
A-4	Provides the distribution of elemental and particulate iodine which originated in the primary containment in the main steamline and drain line piping. Shown is the:

- Fraction of total core inventory on the pipe surfaces as a function of time. (FRACTION IN PIPES)
- Fraction of total core inventory converted to organic iodine which was originally fixed to the pipes and resuspended as a function of:
  - Time integrated release to the condenser.
  - Time integrated release from condenser.

A-5	Provides fraction of core inventory released to environment.

IODINE INVENTORY IN PRIMARY CONTAINMENT
AS A FRACTION OF CORE INVENTORY

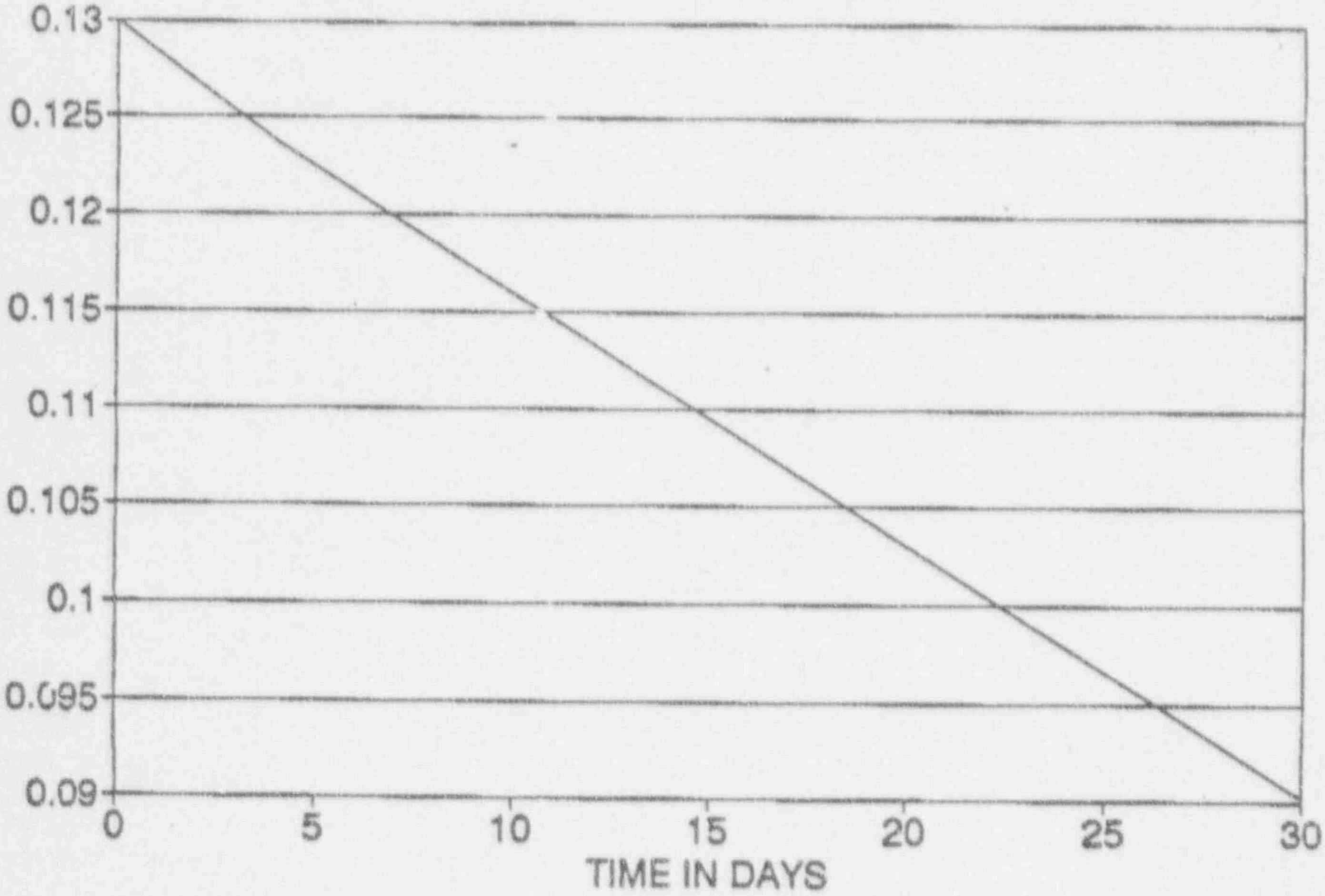


FIGURE A-1

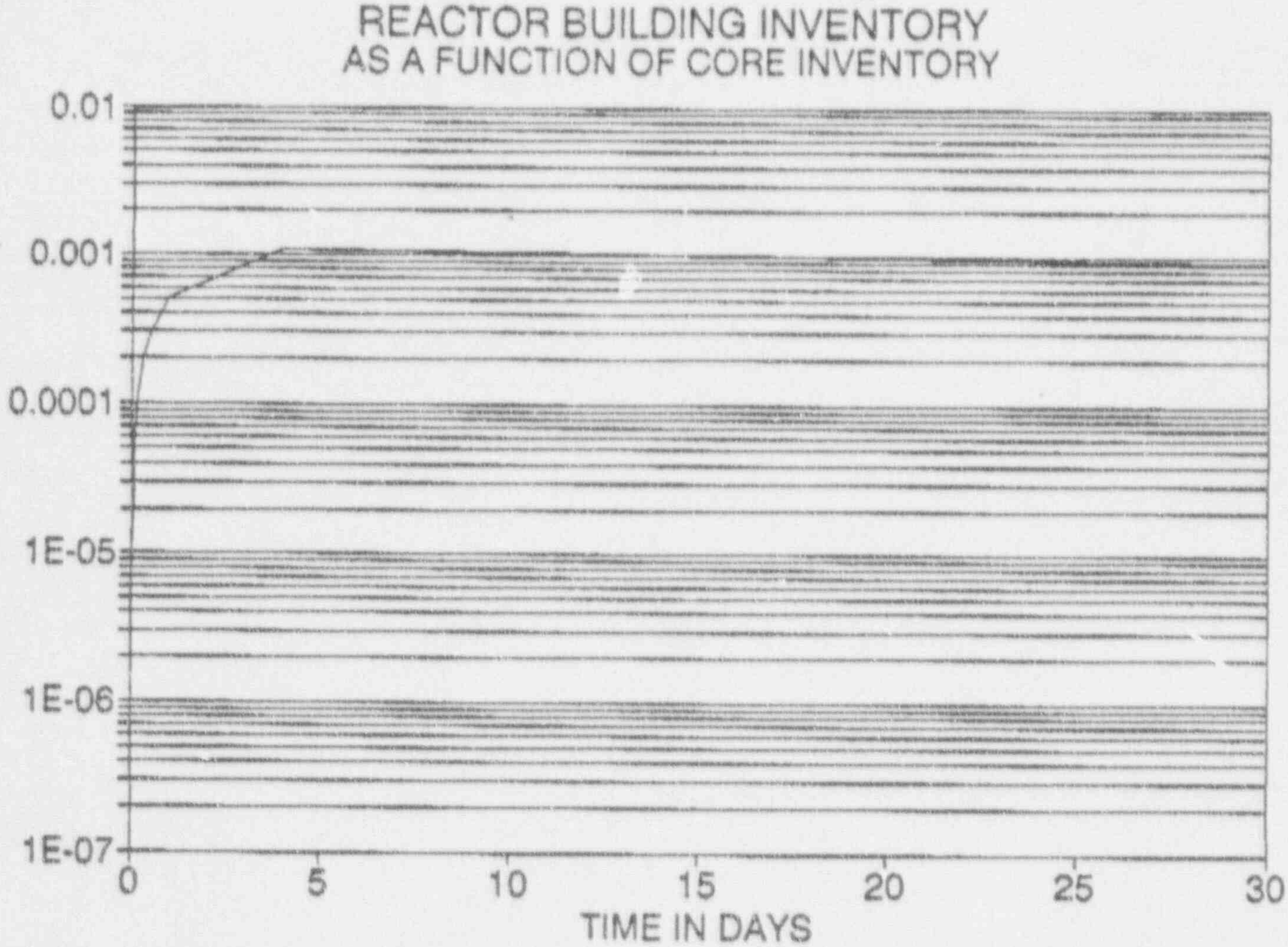

REACTOR BUILDING INVENTORY
AS A FUNCTION OF CORE INVENTORY
0.01
0.001
0.0001
1E-05
1E-06
1E-07
0
5
10
15
20
25
30
TIME IN DAYS

FIGURE A-2

CONDENSER INVENTORY FROM PRIMARY CNTMT
AS A FUNCTION OF CORE INVENTORY

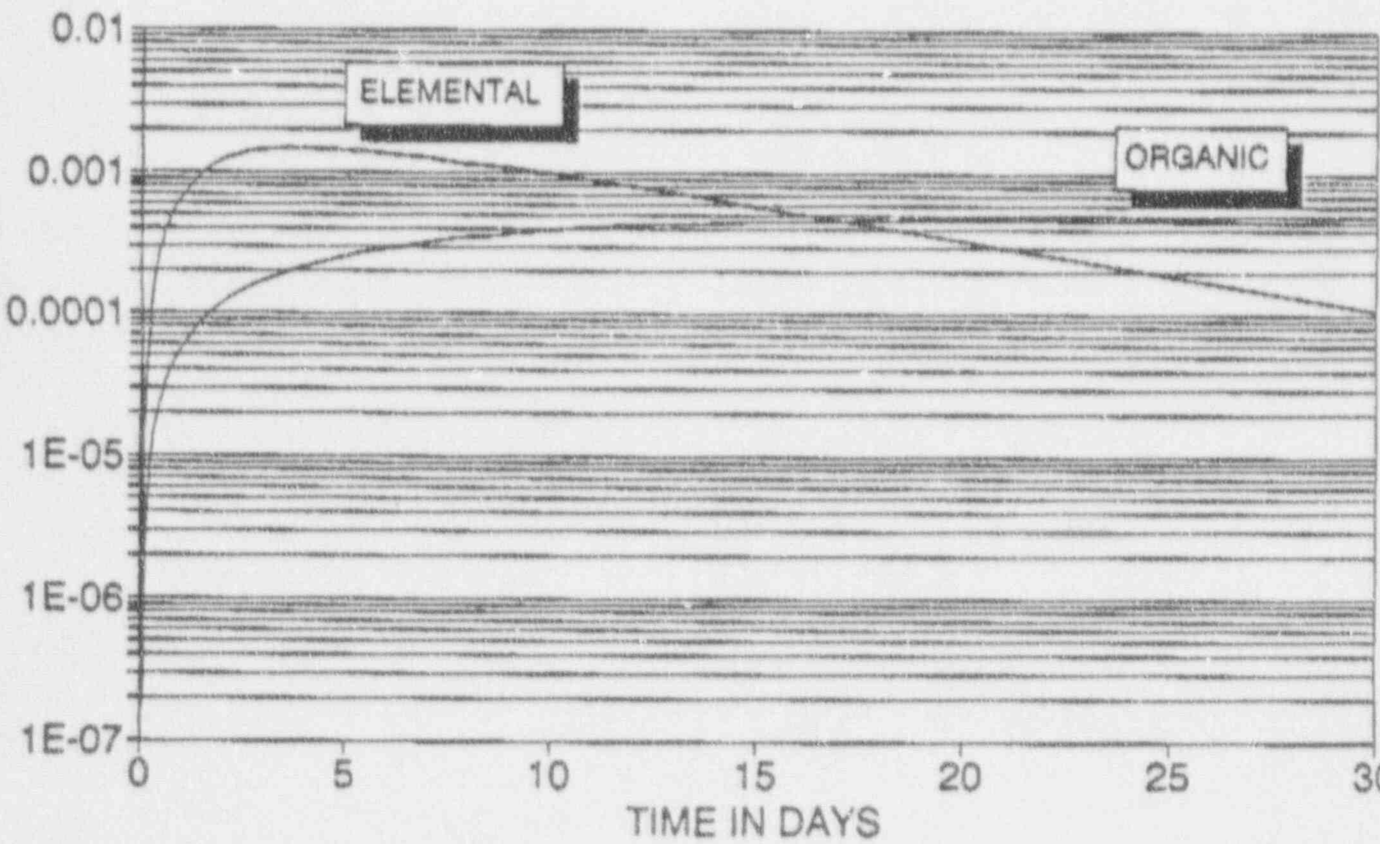


FIGURE A-3

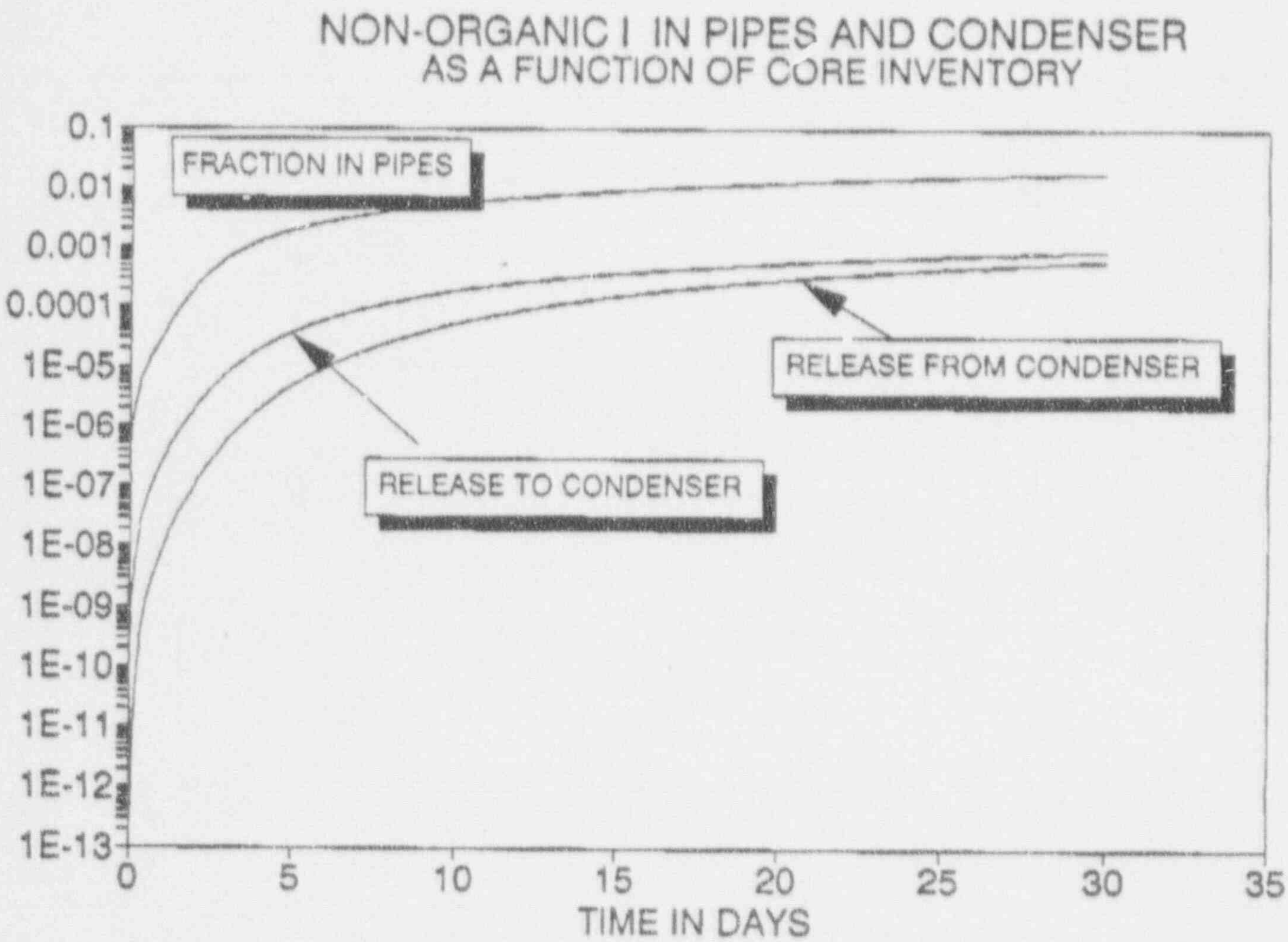

NON-ORGANIC I IN PIPES AND CONDENSER
AS A FUNCTION OF CORE INVENTORY
FRACTION IN PIPES
RELEASE FROM CONDENSER
RELEASE TO CONDENSER
0.1
0.01
0.001
0.0001
1E-05
1E-06
1E-07
1E-08
1E-09
1E-10
1E-11
1E-12
1E-13
0
5
10
15
20
25
30
35
TIME IN DAYS

FIGURE A-4

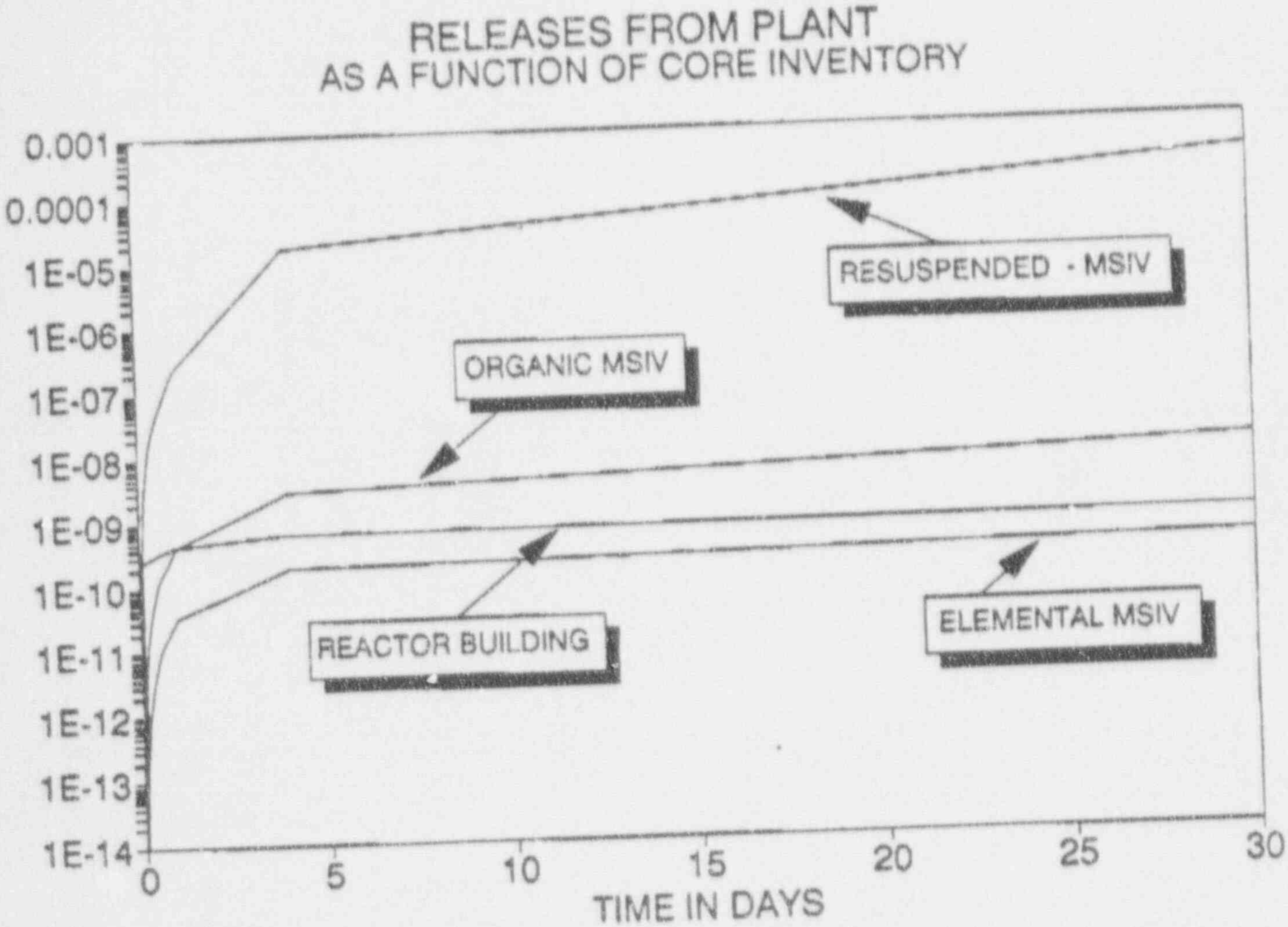

RELEASES FROM PLANT
AS A FUNCTION OF CORE INVENTORY
0.001
0.0001
1E-05
1E-06
1E-07
1E-08
1E-09
1E-10
1E-11
1E-12
1E-13
1E-14
0
5
10
15
20
25
30
TIME IN DAYS
RESUSPENDED - MSIV
ORGANIC MSIV
REACTOR BUILDING
ELEMENTAL MSIV

FIGURE A-5