Proposed Technical Specification Revised Pages

LIST OF FIGURES

3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

3.1.4.1 LIMITING CONDITION FOR OPERATION

The specific activity of the primary coolant shall be limited to:

a. Less than or equal to the most restrictive DOSE EQUIVALENT I-131 limit specified on Figure 3.1-2b, and

b. Less than or equal to 100/ $\bar{E}$ microcuries/gram*

3.1.4.2 APPLICABILITY: at all times except refueling

3.1.4.3 ACTION:

MODES: Power Operation, Start-Up, Hot Standby

a. With the specific activity of the primary coolant greater than the most restrictive DOSE EQUIVALENT I-131 limit specified on Figure 3.1-2b, for more than 48 hours** during one continuous time interval or exceeding the limit line shown on Figure 3.1-2a, be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when DOSE EQUIVALENT I-131 is below the most restrictive limit specified on Figure 3.1-2b.

b. With the specific activity of the primary coolant greater than 100/ $\bar{E}$ microcuries/gram be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when primary coolant activity is less than 100/ $\bar{E}$ microcuries/gram.

MODES: At all times except refueling.

c. With the specific activity of the primary coolant greater than the most restrictive DOSE EQUIVALENT I-131 limit specified on Figure 3.1-2b, or greater than 100/ $\bar{E}$ microcuries/gram perform the sampling and analysis requirements of Table 4.1-3 until the specific activity of the primary coolant is restored to within its limits.

Bases

The maximum limitation on the specific activity of the primary coolant of 1.0 microcurie/gram DOSE EQUIVALENT I-131 ensures that the resulting 2 hour dose at the site boundary will be well within the Part 100 limit following a steam generator tube rupture accident. The limitations on the specific activity of the primary coolant specified on Figure 3.1-2b ensure that the resulting 2 hour doses at the exclusion area boundary and the 30 day LPZ doses will be a small fraction of the Part 100 limit following steam line break accident with postulated accident induced steam generator tube leakage in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM (Figure 3.1-2b represents total leakage from all sources). The limits specified in Figure 3.1-2b represent projected 2-hour leakage and total accident duration leakage based on OTSG tube inspection results, not to exceed 7898 gallons in 2 hours or 24,243 gallons for the accident duration. The allowable activity level is determined based on the OTSG tube inspection results for both 2-hour postulated leakage and total accident duration leakage. The most restrictive primary coolant activity level establishes the maximum limit for the operating cycle.

* $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

** The time period begins from the time the sample is taken.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than the limit specified in 3.1.4.1.a, but within the allowable limit shown on Figure 3.1-2a, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Proceeding to HOT SHUTDOWN prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

The NRC staff has performed a generic analysis of airborne radiation released via the Reactor Building Purge Isolation Valves. The dose contribution due to the radiation contained in the air and steam released through the purge isolation valves prior to closure was found to be acceptable provided that the requirements of Specifications 3.1.4.1, 3.1.4.2 and 3.1.4.3 are met.

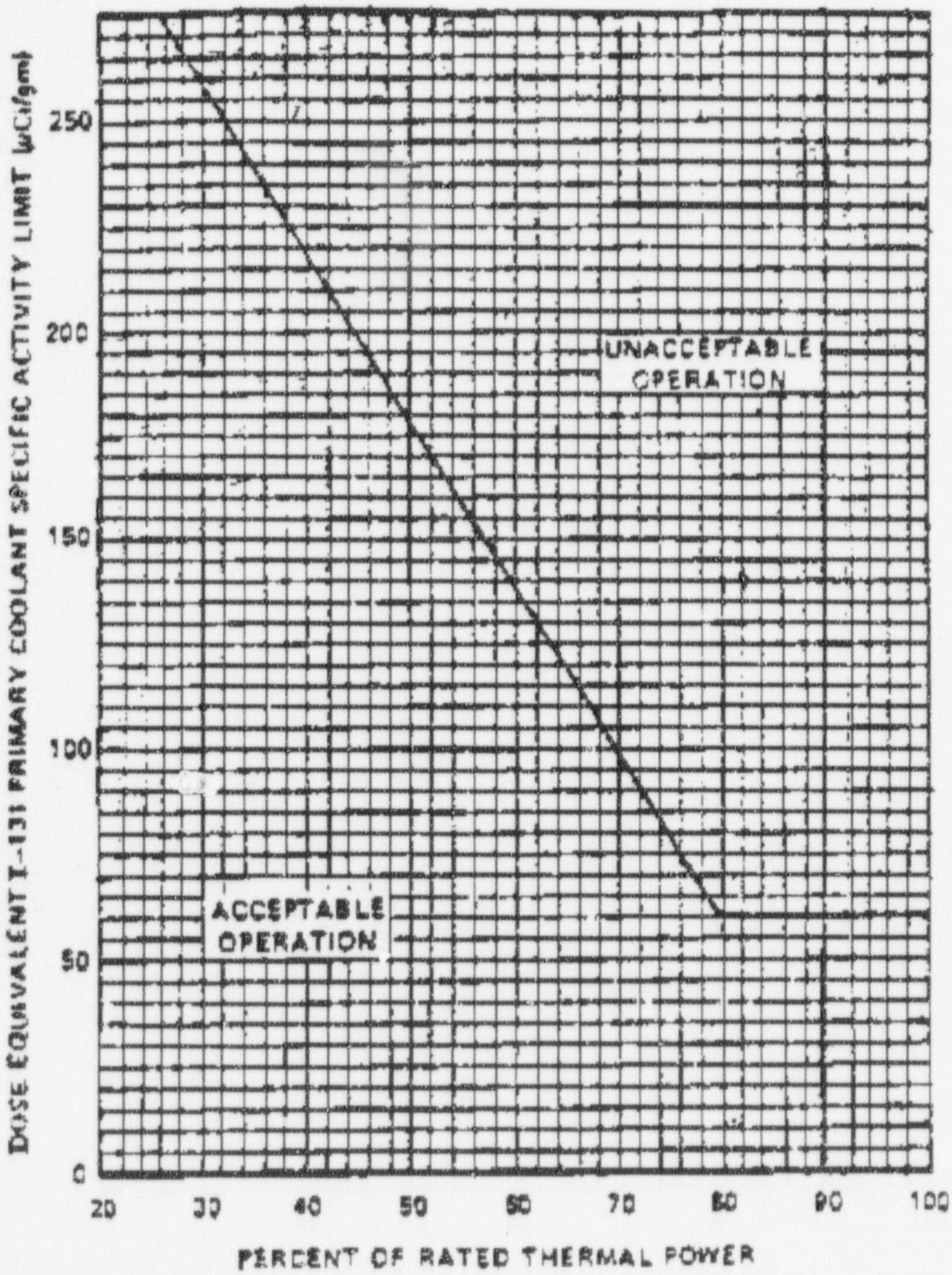
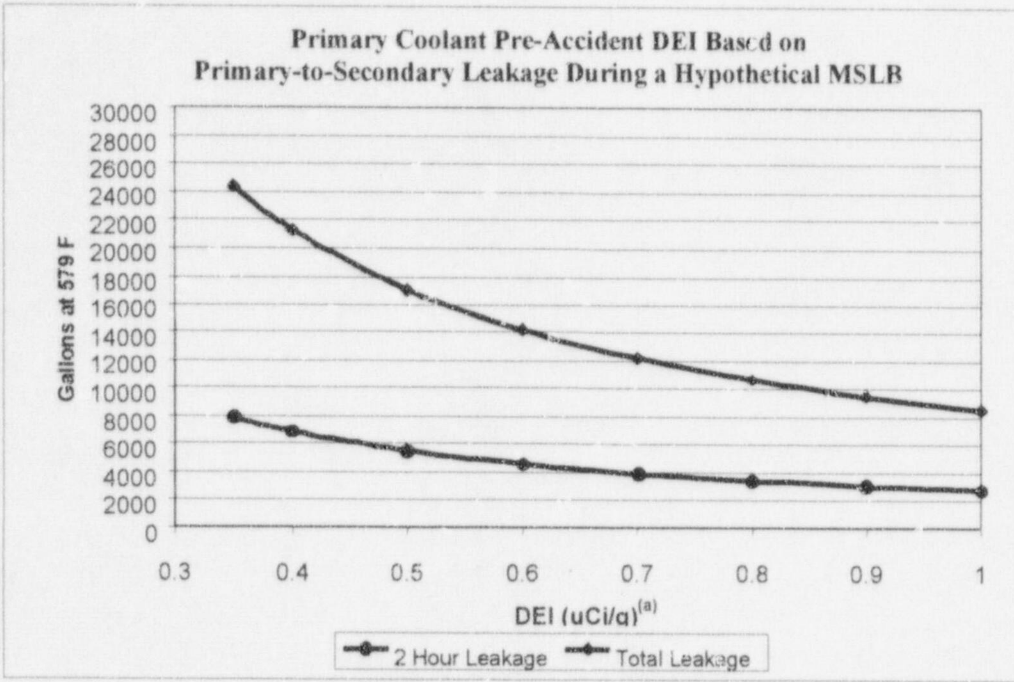


FIGURE 3.1-2a
Dose equivalent I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER



(a) Total iodine activity as dose equivalent I-131

Figure 3.1-2b

Primary Coolant Pre-Accident DEI Based on
Primary-to-Secondary Leakage During a Hypothetical MSLB

TABLE 4.1-3
MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Specific Activity Determination to compare to the 100/ $\bar{E}$ μCi/gm limit	At least once each 72 hours during POWER OPERATION, HOT STANDBY, START-UP, and HOT SHUTDOWN.
	b. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	i) 1 per 14 days during power operations. ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during power operation, start-up and hot standby. iii) #Once per 4 hours, whenever the specific activity exceeds the most restrictive DOSE EQUIVALENT I-131 limi' specified in Figure 3.1-2b, or 100/ $\bar{E}$ μCi/gram during all modes but refueling.
	c. Radiochemical for $\bar{E}$ Determination	1 per 6 months* during power operation.
	d. Chemistry (Cl, F and 0_2)	5 times/week when T_{avg} is greater than 200°F.
	e. Boron concentration	2 times/week
	f. Tritium Radioactivity	Monthly
2. Borated Water Storage Tank Water Sample	Boron concentration	Weekly and after each makeup when reactor coolant system pressure is greater than 300 psig or T_{avg} is greater than 200°F.
3. Core Flooding Tank Water Sample	Boron concentration	Monthly and after each make up when RCS pressure is greater than 700 psig.

ENCLOSURE 1

TMI-1 Technical Specification Change Request No. 272 Safety Evaluation
No Significant Hazards Consideration and
Proposed Technical Specification Revised Pages

I. Technical Specification Change Request No. 272

GPU Nuclear requests that the following changed replacement pages be inserted into existing Technical Specifications:

Revised Technical Specification Pages: vii, 3-8, 3-9, 3-9b, 3-9c, 4-9

These pages are attached to this enclosure.

II. Reason for Change

The purpose of this Technical Specification Change Request is to revise the TMI-1 Technical Specification limit on reactor coolant system specific activity contained in Technical Specification Section 3.1.4 and Table 4.1-3. The proposed change revises the reactor coolant system specific activity limit from 0.35 microcurie/gram dose equivalent iodine I-131 to a cycle specific dependent value based on the once-through steam generator (OTSG) inspection results performed each refueling outage, with a maximum allowable limit of 1.0 microcurie/gram dose equivalent iodine I-131. The proposed maximum allowable limit of 1.0 microcurie/gram (μCi/gm) dose equivalent iodine I-131 was the TMI-1 Technical Specification limit prior to Cycle 12 startup (October 1997).

III. Safety Evaluation Justifying Change

TMI-1 Technical Specification Amendment No. 204 issued October 2, 1997 revised the TMI-1 Technical Specifications to decrease the maximum allowable dose equivalent iodine (I-131) limit in the reactor primary coolant from 1.0 μCi/gm to 0.35 μCi/gm. This revision was required to support a proposed main steam line break (MSLB) accident reanalysis for TMI-1 which accounted for additional accident dose consequences resulting from postulated post-accident once-through steam generator (OTSG) tube leakage. This MSLB accident reanalysis was originally submitted to NRC for review and approval on August 14, 1997. NRC approval of the postulated OTSG post-accident tube leakage analysis was based on NRC's dose analysis which assumed a primary coolant activity limit of 0.35 μCi/gm. This assumption ensured that the resulting dose consequences in the NRC's confirmatory MSLB reanalysis did not exceed a small fraction of the 10CFR Part 100 guidelines or GDC 19 control room operator dose limits. The proposed changes provide a cycle specific primary coolant specific activity limit based on the OTSG inspection results performed each refueling outage, with a maximum allowable limit of 1.0 μCi/gm. The proposed limits are based on an updated reanalysis of the TMI-1 MSLB accident using revised assumptions for atmospheric dispersion coefficients and flashing fraction from the postulated tube leak pathway. The dose consequences remain a small fraction of the 10CFR Part 100 guidelines and GDC-19 limits as described below.

Main Steam Line Break (MSLB) Description

The postulated MSLB is assumed to be the result of a double-ended rupture of a 24 inch outside diameter steam line on one steam generator from 100% power consistent with the existing TMI-1 Steam Line Break design basis accident described in UFSAR Section 14.1.2.9. This is the largest possible break which results in the maximum cooldown rate.

Since the once-through steam generator (OTSG) design has the maximum inventory at full power conditions, starting the event from full power maximizes the heat removal capability of the steam generator. In addition, the postulated MSLB is assumed to occur with the failure of the feedwater regulating valve to the affected steam generator. This is the worst single failure, as it maximizes the overcooling for this event by maximizing the main feedwater flow to the affected generator. The effect of maximizing the overcooling is to maximize the steam generator axial tensile tube loads, which results in the maximum leakage.

Primary-to-secondary leakage from postulated flaws in the kinetically expanded tube-to-tubesheet joint are assumed to occur during the postulated steam line break accident. This total leakage is limited to 2763 gallons in the first two hours, and a total leakage of 8525 gallons over the duration of the accident. Currently approved values of accident induced leakage described in TMI-1 UFSAR Section 14.1.2.9 are a maximum of 3228 gallons for the first 2 hours and 9960 gallons for the duration of the accident. Reactor coolant leakage into the steam generator continues until the Reactor Coolant System (RCS) can be cooled down and the leakage terminated. This was calculated to take a total of 23.33 hours (Reference GPU Nuclear Calculation C-1101-224-E610-060, Rev. 0), and resulted in an average primary-to-secondary leak rate of 23.03 gpm (hot) for the initial 2 hours of the accident, and an average leak rate of 4.50 gpm (hot) for the remaining 21.33 hours of the transient. The basis for the previously approved leakage values is contained in NRC Safety Evaluation Report, dated October 2, 1997, issued for TMI-1 License Amendment No. 204. This accident analysis is performed using Standard Review Plan (SRP)15.1.5, Appendix A assumptions.

GPU Nuclear submittal to the NRC, dated November 26, 1997 (6710-97-2441), provided the methodology used to evaluate the total accident induced primary-to-secondary leakage from the kinetic expansion region that may be postulated to occur during a design basis MSLB. Plant operating and surveillance procedure controls will ensure proper implementation of the cycle specific primary coolant activity level based on the projected MSLB accident-induced primary-to-secondary leak rate.

Iodine Spiking

The environmental consequences from this accident are performed using SRP 15.1.5, Appendix A assumptions (Reference GPU Nuclear Calculation C-1101-900-E000-065, Rev. 0). Iodine spiking effects are analyzed for both an accident-induced spike (AIS) and a pre-accident spike (PAS). For the AIS, it is assumed that the unit has been operating with 1.0 μCi/gm Dose Equivalent Iodine (DEI). The relative isotopic distribution in the reactor coolant is assumed to be the same as Table 14.2-4 of the existing TMI-1 UFSAR, and is adjusted to a mix equivalent to 1.0 μCi/g DEI as shown on Table 1, attached. It is assumed that the reactor trip and/or primary system depressurization creates an iodine spike in the primary system to a value 500 times greater than the release rate corresponding to the 1.0 μCi/g DEI equilibrium activity assumed prior to the accident. No spiking of noble gases is assumed to occur.

For the pre-accident spike it is assumed that a reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to 60 μCi/gm DEI, which is the maximum permitted at 100% power by TMI-1 Technical Specification 3.1.4.3 (operation at this level is not allowed for more than 48 hours). The noble gas concentrations in the RCS were assumed to be at the maximum limit of 100/E-Bar specified by Technical Specification 3.1.4.1 (specific activity at this level requires hot shutdown within 6 hours). Adjustment of Table 14.2-4 of the TMI-1 UFSAR results in isotopic distribution shown on Table 2, attached.

Iodine Partitioning

The amount of Iodine (I_2) that is released from the RCS leakage to the secondary side and ultimately to the environment is calculated in Enclosure 3, Proprietary Polestar Calculation No. PSAT 05653A.04, Rev. 1. This is based on the following considerations for the transport of radioiodine into and through the OTSG:

- I_2 appearance as the result of partitioning as liquid flashes. This occurs due to the elemental iodine in the liquid which partitions to the gas phase.

- I_2 appearance as the result of evaporation-to-dryness of unflashed liquid.

- I_2 appearance as the result of stripping of liquid remaining in the steam generator.

- I_2 deposition in the steam generator, primarily on the Inconel-600 tubes.

A decontamination factor (DF) for I_2 can be calculated based on the pH and deposition of I_2 on the OTSG tubes. The results of the decontamination factor (DF) calculations for radioiodine (as I_2) are shown on the attached Figures 1 and 2 for the AIS and PAS cases respectively. Each Figure shows the RCS pressure and temperature, the flashing fraction, and the total release fraction as functions of time. Figure 1 shows that for the AIS the total release fraction briefly exceeds 25% between about 7000 seconds (about 2 hours) and about 30000 seconds (about 8.5 hours), with an average of about 15% over the 2 –23.33 hour period. For conservatism, a release fraction of 50% will be used for the first 10 minutes and 25% from 10 minutes to 23.33 hours. Figure 2 shows that for the PAS, the total release fraction exceeds 25% for only the first 10 minutes then decreases over the duration of the transient. The flashing fraction decreases from about 45% to about 25% in the first 10 minutes of the transient, and then further decreases approximately linearly to near zero at 23.33 hours. For conservatism, a release fraction of 50% will be used for the first 10 minutes and 25% from 10 minutes to 23.33 hours. These assumed release fraction values are greater than the flashing fraction for both cases.

Radiological Analysis

The STARDOSE computer program is used to calculate the control room and offsite dose consequences. STARDOSE is proprietary to Polestar Applied Technology and is fully consistent with applicable Code of Federal Regulations and Regulatory Guides. The computer program is prepared under the Polestar Quality Assurance Program and intended for use in applications covered by Appendix B to 10 CFR 50 and 10 CFR 21. A STARDOSE model schematic is shown in attached Figure 3.

The TMI-1 control room habitability evaluation methodology and assumptions, including atmospheric dispersion factors for releases to the control room, are described in GPU Nuclear's letter to the NRC dated March 24, 1998. The dispersion coefficients calculated for the Maximum Hypothetical Accident in the above referenced letter are conservatively used for the MSLB evaluation. The releases occurring from the MSLB accident will be diffused first through the Intermediate Building prior to being released to the environment. Any release from the Intermediate Building will be impacted by the total building complex blockage area including reactor building, turbine building, service building and intermediate building. Released activity will disperse rapidly into the building complex wake. Site geometry shows that distances to the control building ventilation system intake (yard intake) are greater for this release than assumed in the MHA analysis (112 meters for Intermediate Building releases compared to 91 meters for reactor building releases). This increased distance will result in lower X/Q values than those calculated if actual distances are utilized. Therefore it is considered a conservative approach to use X/Q values from the MHA since actual values will be lower due to increased blockage, initial building diffusion, and increased distance to the yard intake.

The atmospheric dispersion coefficients utilized for the UFSAR Chapter 14 MSLB accident reanalysis are determined on a directional dependent basis with fixed Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) boundary as indicated in existing TMI-1 Technical Specification Section 5.1.1. Previous atmospheric dispersion analysis assumed a variable EAB boundary.

TMI-1 Technical Specification Section 5.1.1 defines the EAB as a 2000 ft. (610m) radius determined from the minimum distance in an easterly direction from the plant to the shore of the mainland. Figure 5-1 in the Technical Specifications indicates the exclusion area as a stretched circle centered equidistant between the TMI-1 and TMI-2 reactor buildings with circular radius centered at each unit's reactor building centerline equal to 2000 ft. This stretched circle configuration defines the EAB from all directions and serves as the basis for determining EAB X/Q values using meteorological data from the site and Regulatory Guide 1.145 methods. The LPZ is defined in Technical Specification 5.1.1 as an area with a two-mile radius and is depicted in Technical Specification Figure 5-2. The X/Q values for the LPZ have also used Regulatory Guide 1.145 methods for determination. The new X/Q values have been determined by subtracting containment radius from the boundary radii in each of the sixteen directions evaluated.

The evaluation of accident X/Q values for TMI-1 has also been updated to reflect recent site meteorological data collected during four (4) years including 1992, 1993, 1995, 1996 with data recovery above 90% and Regulatory Guide 1.145 methodology. This data is provided in Attachment 1 and is the latest available data. Less than 90% of the 1994 data was recovered, therefore, this year was excluded. The proposed accident X/Q values are based on a larger data base of meteorological data than the current TMI-1 UFSAR accident X/Q values. This larger and updated database provides a more accurate representation of conservative accident X/Q values than previously utilized in design basis accident analysis. Data used in the analysis were collected on the site meteorological tower located at the northern end of Three Mile Island. The data used were hourly values of speed and direction from the 100 ft. level and vertical temperature difference between 150 ft. and 33 ft. Since wind speed measurements at the site are made at the 100 ft. level, they were adjusted to the standard 10 meter (33 ft.) level utilizing a power law relationship as a function of stability. The Pasquill diffusion class was determined using vertical temperature difference (delta-T) and the categories given in NRC Regulatory Guide 1.23. Values of σ_Y and σ_Z used in the Regulatory Guide 1.145 dispersion equations were determined as a function of distance and stability class using the standard Pasquill-Gifford curves (Figures 1 and 2 of Regulatory Guide 1.145).

Pickard, Lowe & Garrick's (PLG) WINDOW code was used to perform the Regulatory Guide 1.145 calculations (Reference PLG report "Accident X/Q Values for TMI-1", dated July 23, 1998). It has been used since 1969 for calculating X/Q values in support of nuclear plant site evaluations. Values of X/Q for the EAB were determined for each hour of the 4-year data base using the Regulatory Guide 1.145 equations. Cumulative probability distributions were made for each of the 16 direction sectors for the 4-year period. An envelope was constructed around all 16 direction-dependent curves and the 0.5% probable value (i.e., the value exceeded no more than 0.5% of the time) was determined to be 8.0E-4 sec/m^3. These results are shown in Figure 4. A second criterion required by the Regulatory Guide 1.145 procedure is that the 5% probable direction-independent X/Q value must be less than the 0.5% value for the direction-dependent case. Inspection of Figure 6 shows the 5% value is 7.8E-4 sec/m^3 which is less than 8.0E-4 sec/m^3 direction-dependent value. According to the regulatory guide, the direction-dependent value must be used since it is higher. Table 4 summarizes the EAB results.

For the LPZ, the Regulatory Guide 1.145 procedure for different averaging times was used. The direction-dependent X/Q values at the LPZ for each direction are shown in Figure 5. As shown in Figure 7, the 1-hour LPZ 0.5% X/Q (4 years) was plotted on log-log graph paper at 2 hours and a straight-line was drawn to the annual average value at 8760 hours. Values at intermediate averaging times were taken from the straight-line connecting the two points. These values can be compared with more realistic 0.5% probable averages computed by the WINDOW code shown as the lower line on Figure 7. This comparison showed that the Regulatory Guide 1.145 technique is conservative by more than a factor of two for intermediate averaging times beyond 8 hours. The sector average equations in Regulatory Guide 1.111 were used after the first 8 hours. The 5% probable X/Q values at the LPZ for each averaging time are plotted on Figure 8. Results are summarized in Table 4.

Radiological Consequences

The radiological consequences of the bounding integrated primary-to-secondary accident induced leakage with a maximum allowable DEI-131 limit of 1.0 μCi/gm are shown on the attached Table 3 for the PAS and AIS cases (Reference GPU Nuclear Calculation C-1101-900-E000-065, Rev. 0). The results show that for an MSLB with an assumed pre-accident iodine spike the calculated doses are well within the guideline values of 10 CFR Part 100. For the MSLB with an assumed accident initiated iodine spike, the calculated doses are slightly higher than the existing UFSAR, but remain a small fraction of the 10 CFR Part 100 limit. For both cases, the resulting 30 day control room doses are well below the 10 CFR Appendix A, GDC 19 limits and the NRC Standard Review Plan 6.4 guideline of 30 Rem, and remain bounded by the March 24, 1998 TMI-1 design basis Control Room Habitability Analysis. The results are also presented graphically on the proposed Technical Specification Figure 3.1-2b, which plots allowable DEI limit based on primary-to-secondary leakage (Reference GPU Nuclear Calculation C-1101-900-E000-071, Rev. 0). The primary-to-secondary leakage values shown represent projected 2-hour leakage and total accident duration leakage based on OTSG tube inspection results. The most restrictive leakage value establishes the maximum primary coolant activity level for the subsequent operating cycle. A higher leakage will result in lower DEI limit, or conversely, a lower leakage will allow a higher DEI up to a maximum allowable limit of 1.0 μCi/gm.

Therefore the proposed change in the TMI-1 Technical Specification required RCS specific activity level does not adversely affect nuclear safety or safe plant operation.

Table 1 – RCS Iodine and Noble Gas Activities for AIS

RCS Iodine Activities at 1 uCi/g DEI					
	FSAR RCS Isotopic (uCi/g) [(a)]	ICRP 30 DCF (Rem/Ci) [(b)]	FSAR RCS DEI (uCi/g) [(c)]	Adjusted FSAR RCS Isotopic (uCi/g) [(d)]	Adjusted FSAR RCS DEI (uCi/g) [(e)]
I-131	5.71E+00	1.08E+06	5.71E+00	**8.37E-01**	8.37E-01
I-132	1.92E+00	6.44E+03	1.14E-02	**2.81E-01**	1.68E-03
I-133	6.07E+00	1.80E+05	1.01E+00	**8.90E-01**	1.48E-01
I-134	7.57E-01	1.07E+03	7.47E-04	**1.11E-01**	1.09E-04
I-135	3.08E+00	3.13E+04	8.92E-02	**4.52E-01**	1.31E-02
Totals	1.75E+01		6.82E+00	**2.57E+00**	1.00E+00

ISOTOPE	FSAR Table 14.2-4 RCS (uCi/g) [(f)]	FSAR Table Adjusted to 1 uCi/g DEI (uCi/g) [(g)]
KR-83M	5.30E-01	7.77E-02
KR-85M	2.43E+00	3.56E-01
KR-85	9.75E+00	1.43E+00
KR-87	1.28E+00	1.88E-01
KR-88	3.95E+00	5.79E-01
XE-131M	2.68E+00	3.93E-01
XE-133M	4.22E+00	6.19E-01
XE-133	3.92E+02	5.75E+01
XE-135M	4.85E-01	7.11E-02
XE-135	8.37E+00	1.23E+00
XE-138	6.92E-01	1.01E-01

(a) Iodine isotopic activities from Table 14.2-4 of the TMI-1 FSAR (1% failed fuel).

(b) Iodine dose conversion factors from ICRP 30.

(c) Iodine isotopic activities Table 14.2-4 of the TMI-1 FSAR converted to dose equivalent Iodine 131 (DEI).

(d) Iodine isotopic activities from Table 14.2-4 of the TMI-1 FSAR scaled down to produce a DEI of 1.0 uCi/g.

(e) Scaled down isotopic activities converted to dose equivalent I-131.

(f) Noble gas isotopic activities from Table 14.2-4 of the TMI-1 FSAR (1% failed fuel).

(g) Noble gas isotopic activities that maintain the same relative abundance as 1% failed fuel but scaled down to coincide with the iodines at 1.0 uCi/g DEI.

Table 2 – RCS Iodine and Noble Gas Activities for PAS

RCS Iodine Activities at 60 uCi/g DEI

	FSAR RCS Isotopic (uCi/g) [(a)]	ICRP 30 DCF (Rem/Ci) [(b)]	FSAR RCS DEI (uCi/g) [(c)]	Adjusted FSAR RCS Isotopic (uCi/g) [(d)]	Adjusted FSAR RCS DEI (uCi/g) [(e)]
I-131	5.71E+00	1.08E+06	5.71E+00	**5.02E+01**	5.02E+01
I-132	1.92E+00	6.44E+03	1.14E-02	**1.69E+01**	1.01E-01
I-133	6.07E+00	1.80E+05	1.01E+00	**5.34E+01**	8.89E+00
I-134	7.57E-01	1.07E+03	7.47E-04	**6.66E+00**	6.57E-03
I-135	3.08E+00	3.13E+04	8.92E-02	**2.71E+01**	7.85E-01
Totals	1.75E+01		6.82E+00	**1.54E+02**	6.00E+01

RCS Noble Gas Activity

	$T_{1/2}$ (min)	C_i FSAR RCS (uCi/g) [(f)]	γ_i Avg Gamma (MeV/dis)	β_i Avg Beta (MeV/dis)	Ebar Calc $(C_i)(\gamma_i+\beta_i)$ [(g)]	100/Ebar RCS (uCi/g) [(h)]
KR-83M	109.8	0.53	0.0026	0.0382	0.021624	0.539
KR-85M	264	2.43	0.152	0.243	0.95985	2.472
KR-85	5658000	9.75	0.00211	0.222	2.1850725	9.917
KR-87	76.2	1.28	1.42	1.05	3.1616	1.302
KR-88	168	3.95	1.74	0.34	8.216	4.018
XE-131M	16992	2.68	0.271	0.137	1.09344	2.726
XE-133M	3252	4.22	0.0567	0.177	0.986214	4.292
XE-133	7590	392	0.0497	0.146	76.7144	398.721
XE-135M	15.6	0.485	0.429	0.098	0.255595	0.493
XE-135	548.4	8.37	0.248	0.316	4.72068	8.513
XE-138	14.4	0.692	N/A	N/A	N/A	0.704
	Total====>	426.387		$\Sigma(C_i)(\gamma_i+\beta_i)$=>	98.314	433.697
				Ebar ===>	0.231	
				100/ebar	433. 7	

(a) Iodine isotopic activities from Table 14.2-4 of the TMI-1 FSAR (1% failed fuel)
(b) Iodine dose conversion factors from ICRP 30
(c) Iodine isotopic activities Table 14.2-4 of the TMI-1 FSAR converted to dose equivalent Iodine 131 (DEI)
(d) Iodine isotopic activities from Table 14.2-4 of the TMI-1 FSAR scaled up to produce a DEI of 60 uCi/g
(e) Scaled up isotopic activities converted to dose equivalent I-131
(f) Noble gas isotopic activities from Table 14.2-4 of the TMI-1 FSAR (1% failed fuel)
(g) Calculation of Ebar to determine 100/Ebar = 433.7 uCi/g for the FSAR mix.
(h) Noble gas isotopic activities scaled up to produce a total activity of 433.7 uCi/g while maintaining the same relative abundance as 1% failed fuel.

Table 3 – MSLB Dose Results

Accident Initiated Spike (AIS) Doses (Rem)

	Thyroid	Whole Body	Skin
Control Room Doses	3.32E+00	5.34E-04	4.11E-03
EAB Doses	2.97E+01	9.96E-02	N/A
LPZ Doses	2.54E+01	4.85E-02	N/A

Note : For the MSLB with an accident initiated spike, the dose acceptance criteria for the EAB and LPZ are 2.5 Rem whole body and 30 Rem thyroid. The criteria for the control room are 5 Rem whole body and 30 Rem thyroid.

Pre-Accident Spike (PAS) Doses (Rem):

	Thyroid	Whole Body	Skin
Control Room Doses	5.02E+00	1.80E-03	1.81E-02
EAB Doses	4.09E+01	1.88E-01	N/A
LPZ Doses	1.20E+01	4.44E-02	N/A

Note : For the MSLB with a pre-accident spike, the dose acceptance criteria for the EAB and LPZ are 25 Rem whole body and 300 Rem thyroid. The criteria for the control room are 5 Rem whole body and 30 Rem thyroid.

TABLE 4

X/Q Calculations For TMI-1 EAB and LPZ

Dose Calculation Time (hours)	Averaging Time (hours)	0.5% X/Q at EAB (sec/m^3)	Direction (Toward)	5% X/Q at EAB (sec/m^3)	0.5% X/Q at LPZ (sec/m^3)	Direction	RG1.145 (sec/m^3)	5% X/Q at LPZ (sec/m^3)
-	1	8.0E-4	N	7.8E-4	1.4E-4	NE	1.4E-4	1.4E-4
0-2	2	5.6E-4	N	-	9.8E-5	NE	1.4E-4	9.4E-5
2-8	8	-	-	-	4.8E-5	N**	6.0E-5*	4.6E-5
8-24	16	-	-	-	8.7E-6	WNW**	3.9E-5*	8.3E-6
24-96	72	-	-	-	4.4E-6	WNW**	1.6E-5*	4.2E-6
96-720	624	-	-	-	2.0E-6	N**	4.0E-6*	1.8E-6
Annual Avg.	8760	-	-	-	-	NE	7.8E-7	-

*Interpolated from Figure 7

**Values in the NE direction would be lower

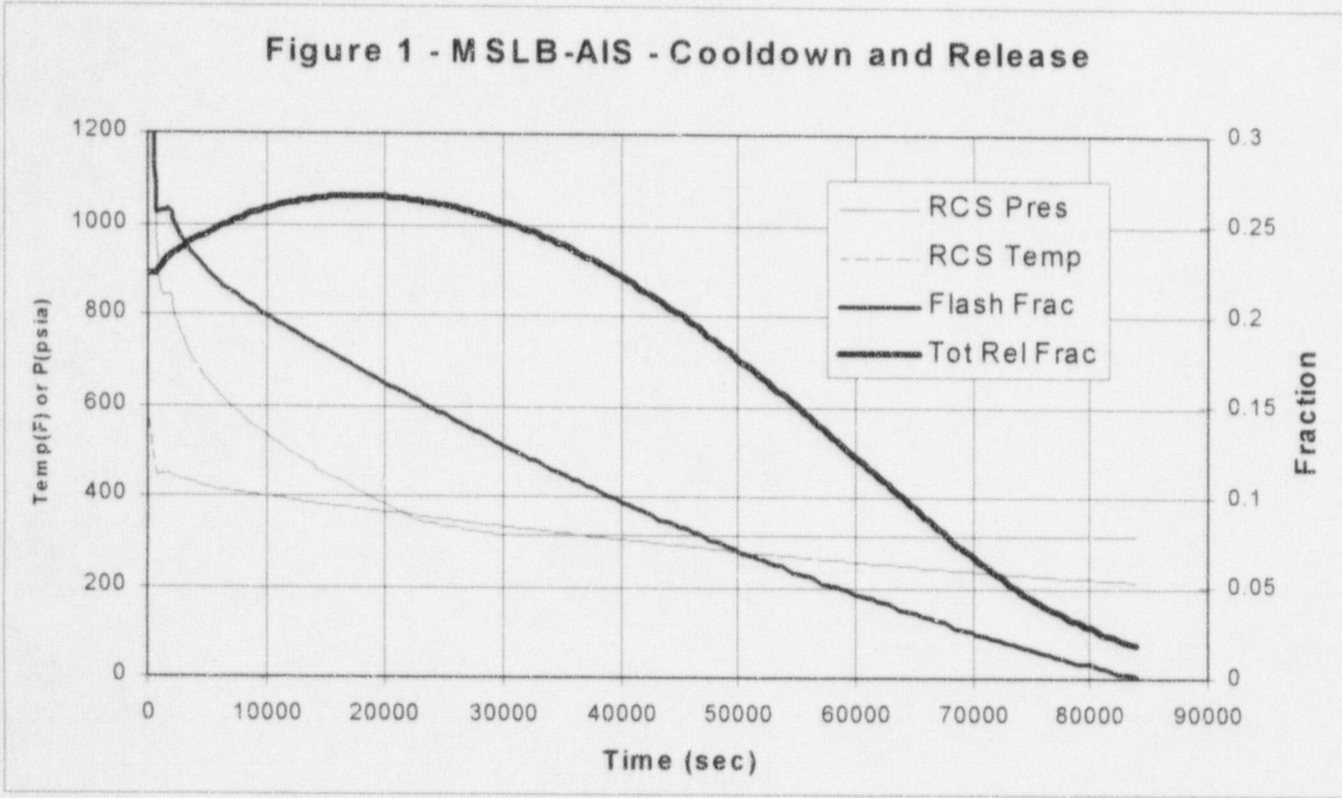


Figure 1 - MSLB-AIS - Cooldown and Release
RCS Pres
RCS Temp
Flash Frac
Tot Rel Frac
Temp(F) or P(psia)
Fraction
Time (sec)
0
200
400
600
800
1000
1200
0.05
0.1
0.15
0.2
0.25
0.3
10000
20000
30000
40000
50000
60000
70000
80000
90000

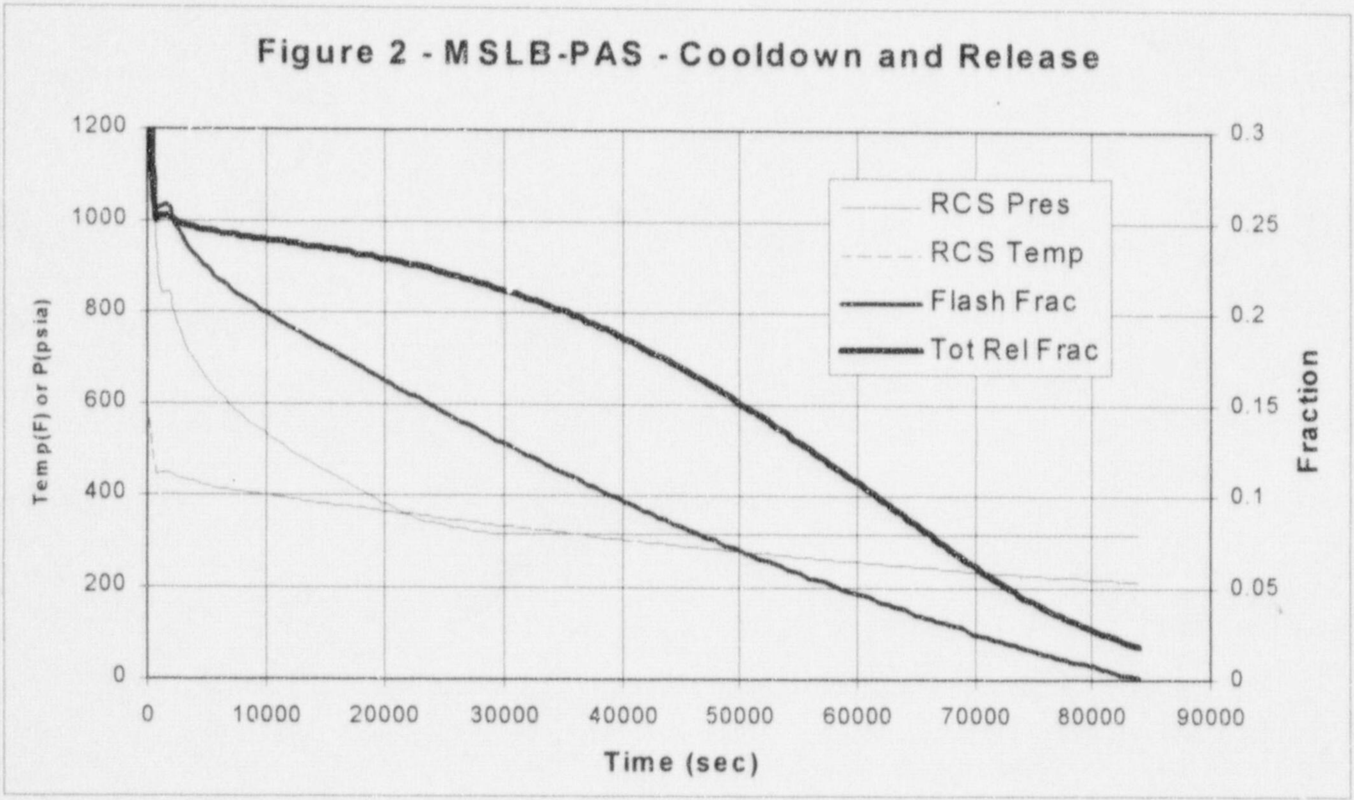
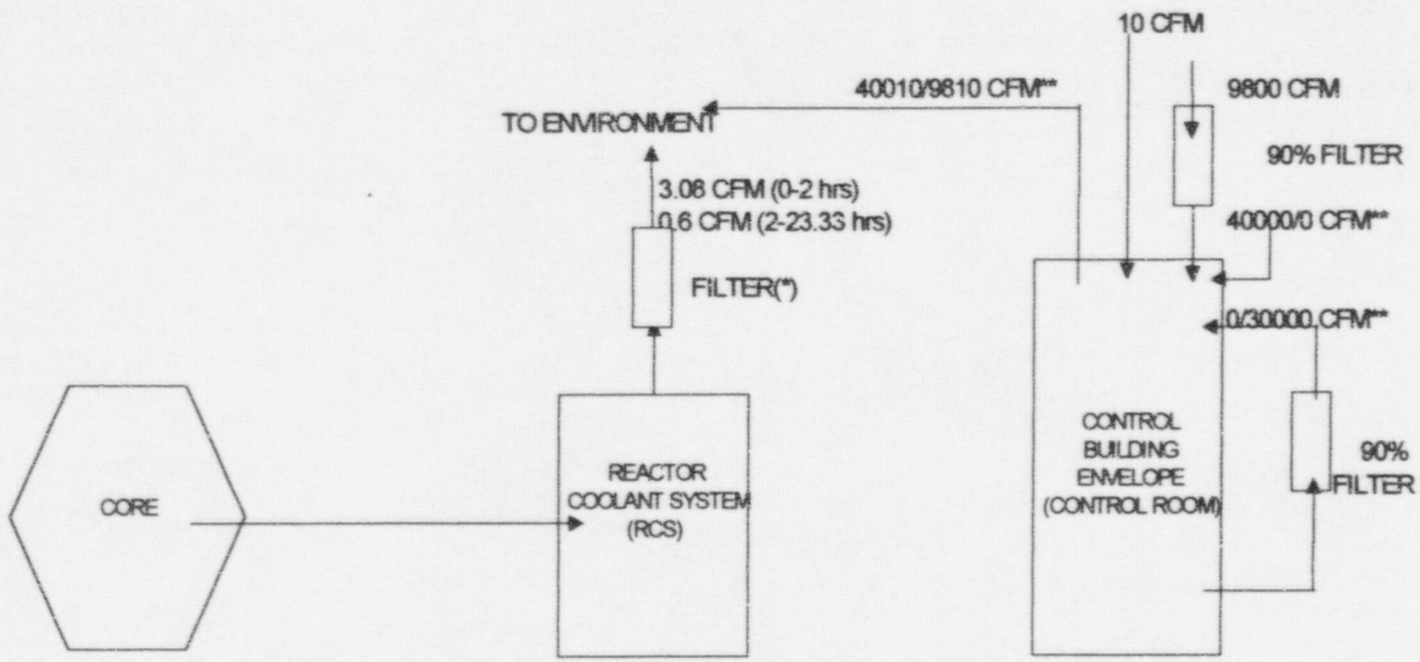


Figure 2 - MSLB-PAS - Cooldown and Release
1200
1000
800
600
400
200
0
Temp(F) or P(psia)
0.3
0.25
0.2
0.15
0.1
0.05
0
Fraction
0
10000
20000
30000
40000
50000
60000
70000
80000
90000
Time (sec)
RCS Pres
RCS Temp
Flash Frac
Tot Rel Frac

Figure 3 - STARDOSE MODEL FOR TMI-1 CONTROL ROOM DOSE ANALYSIS - MSLB



* For AIS - release fraction is 0.5 (0-10 mts) and 0.25 (10 mts - 23.33 hrs)
For PAS - release fraction is 0.5 (0-10 mts) and 0.25 (10 mts - 23.33 hrs)

** First value is the flow rate for the first 1 hour prior to manual action to place system in recirculation.

Figure 4 - X/Q Values Based on 4 Years of TMI Data (92, 93, 95, 96)

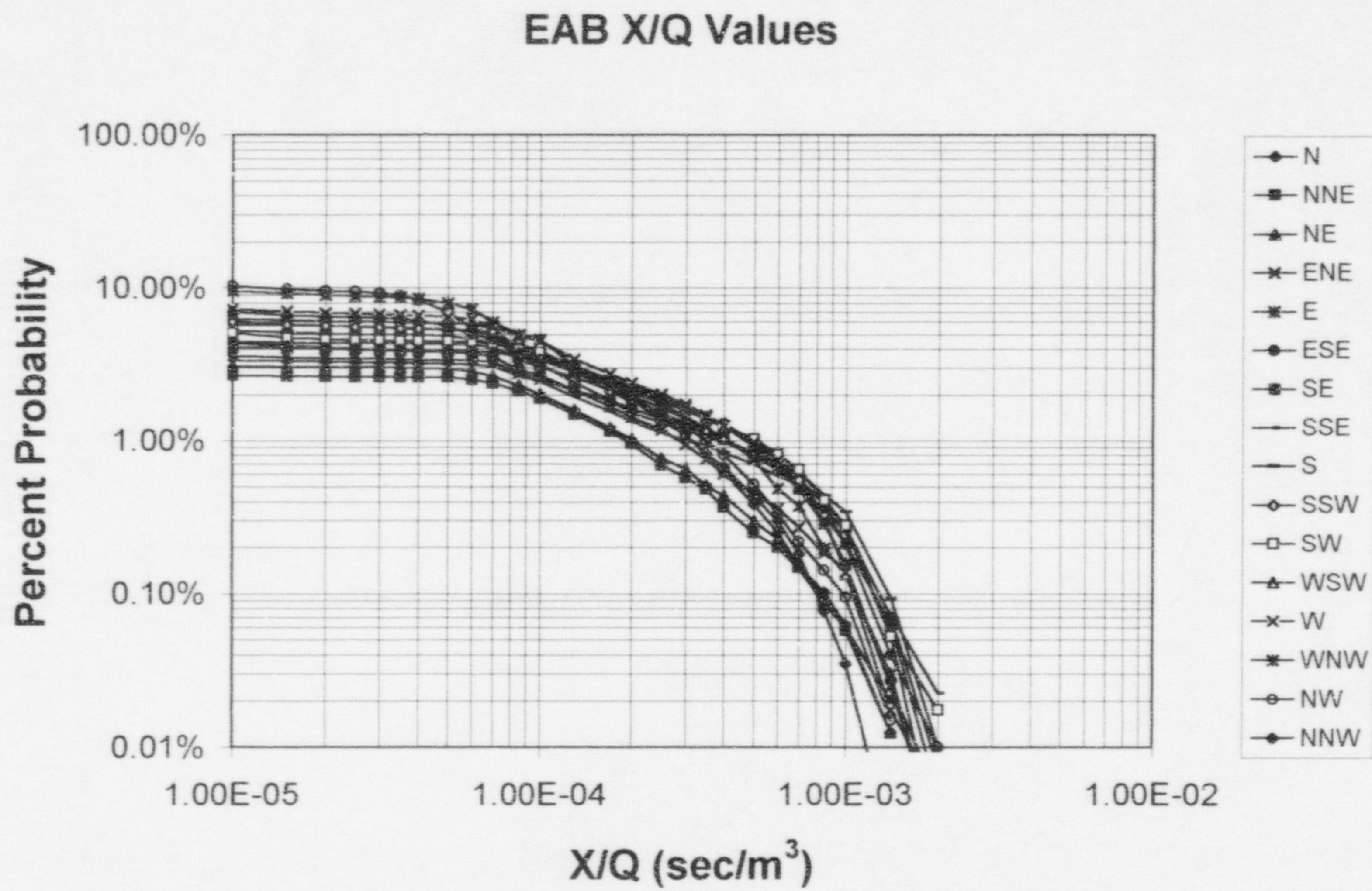


Figure 5 - X/Q Values Based on 4 Years of TMI Data (92, 93, 95, 96)

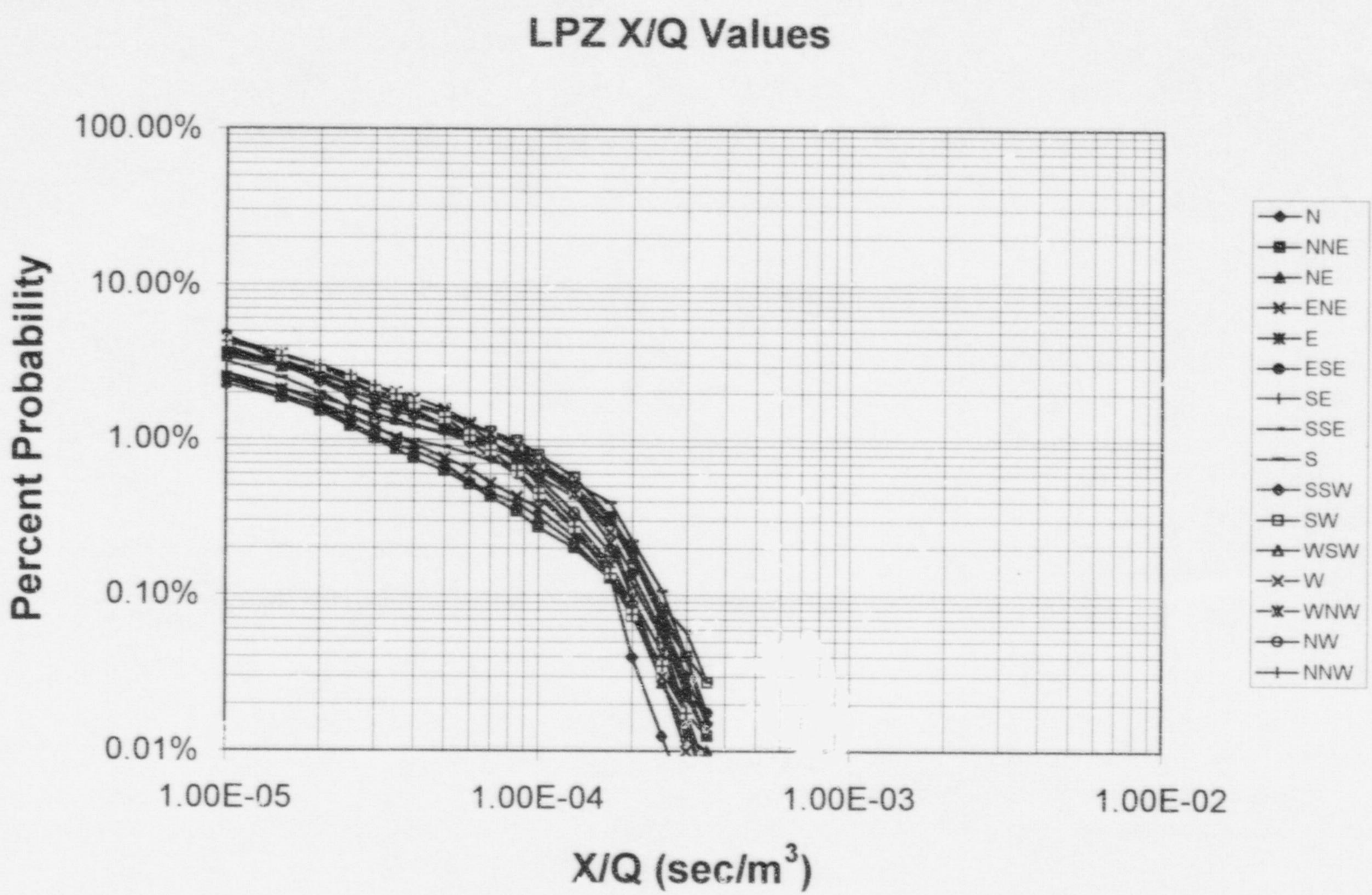


Figure 6 - 5% Probable 1 Hour Direction Independent X/Q at the EAB (92, 93, 95, 96 Data)

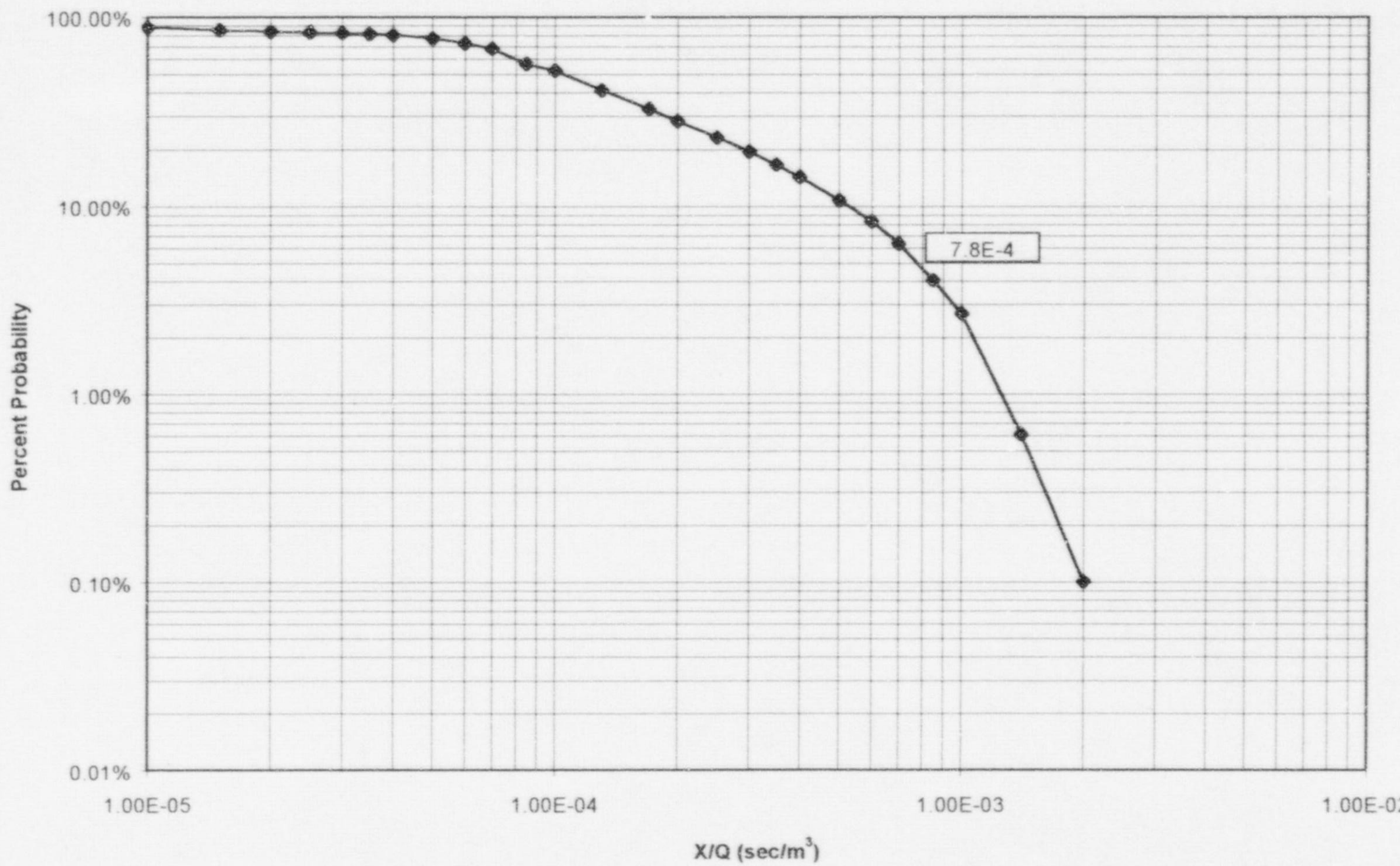


Figure 7 - LPZ X/Q Values Based on 4 Years of TMI Data (92, 93, 95, 96)

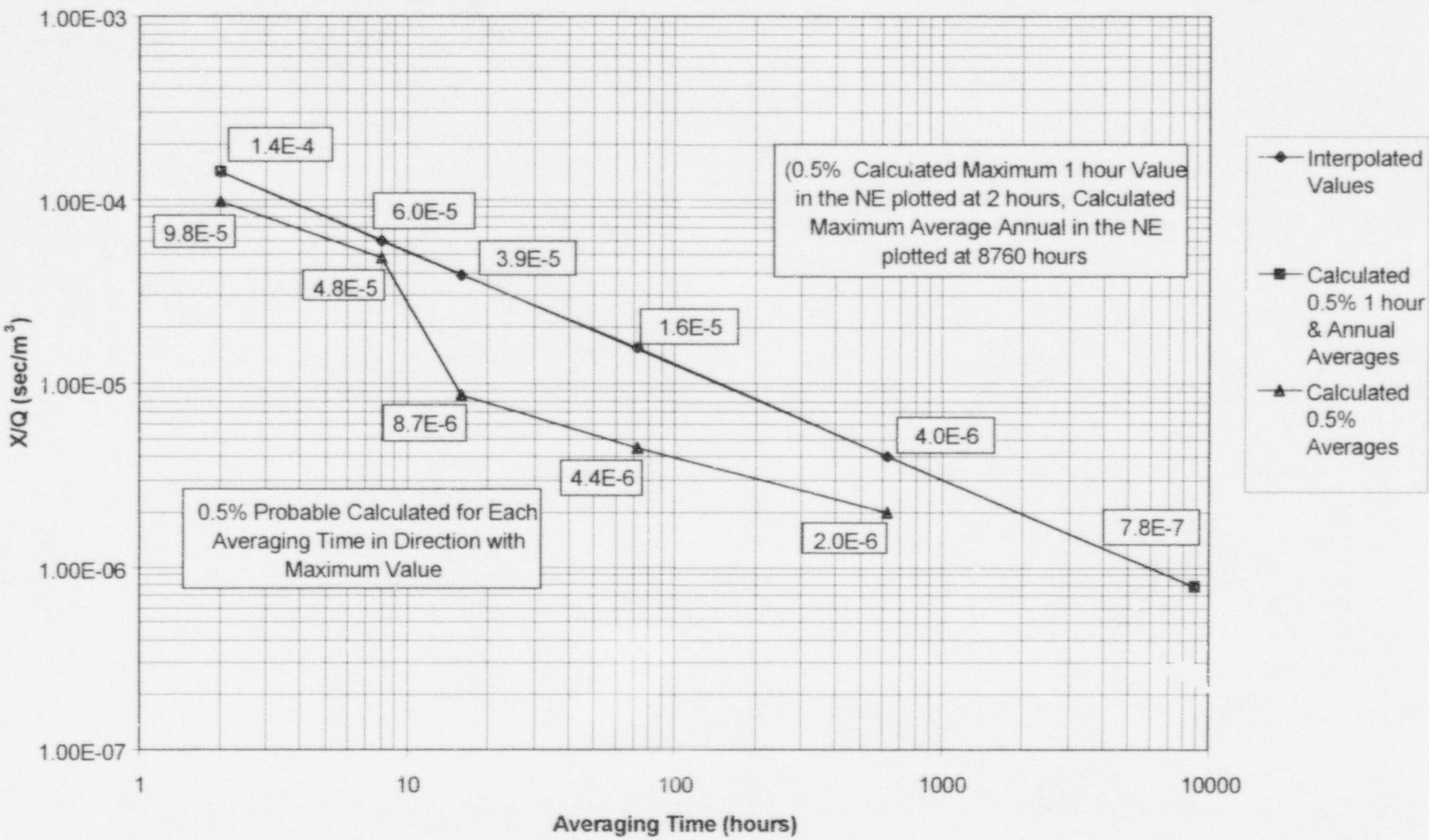
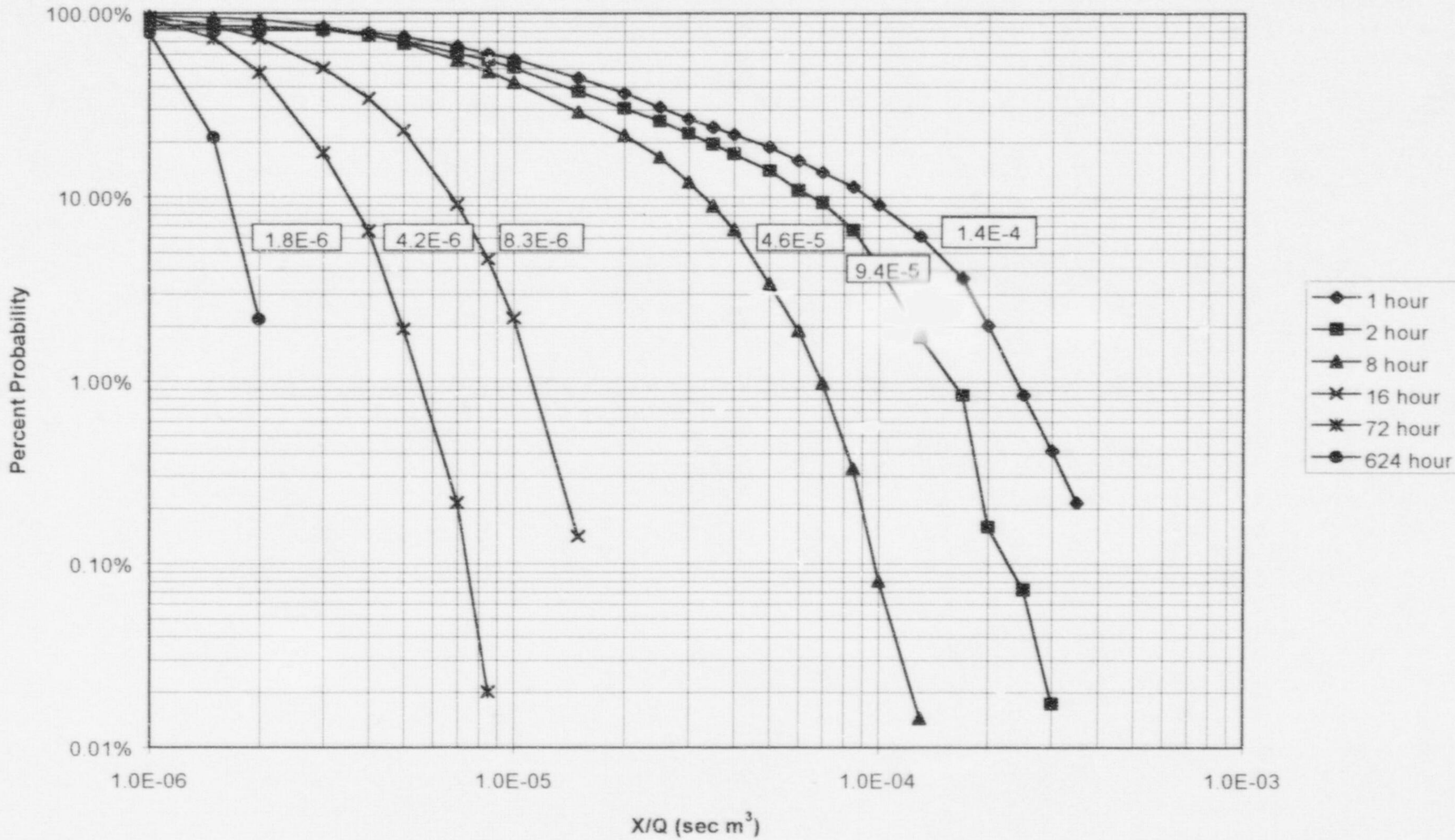


Figure 8 - 5% Probable Direction Independent X/Q at the LPZ (92, 93, 95, 96 Data)



IV. No Significant Hazards Consideration

GPU Nuclear has determined that this Technical Specification Change Request poses no significant hazards consideration as defined by 10CFR50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated. The proposed amendment has no affect on structures, systems or components. The existing steam line break criteria are maintained. This change only accounts for radiological consequences resulting from a revised maximum allowable reactor coolant system (RCS) specific activity limit of 1.0 μCi/gm. The new radiological consequences of the revised MSLB accident, which also incorporate more conservative values for atmospheric dispersion, are below 10CFR100 limits and 10CFR50, Appendix A, GDC-19 limits for the control room. The use of revised atmospheric dispersion factors for other TMI-1 accident analyses is addressed in a separate license amendment request submittal. Therefore, this activity does not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any previously evaluated. The proposed amendment has no impact on any plant structures, systems or components OTSG tube structural integrity is maintained. Therefore, this activity does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment has no impact on structures, systems, or components. OTSG tube inspection criteria are not affected by this change. OTSG tube structural integrity is maintained. The existing TMI-1 Technical Specification Section 3.1.4.1 Bases state that the limitations on the specific activity of the primary coolant ensure that the resulting 2-hour doses at the site boundary will be well within the Part 100 limit following associated design basis accidents postulated in conjunction with an assumed steady state primary-to-secondary steam generator tube leakage of 1.0 gpm. This margin of safety is preserved since resulting dose consequences incorporating more conservative values for atmospheric dispersion remain well within the Part 100 limit. Therefore, this activity does not reduce the margin of safety.

V. Implementation

GPU Nuclear requests that the amendment authorizing this change become effective immediately upon issuance.

ATTACHMENT 1

1992, 1993, 1995, 1996 Meteorological Data
(Supplied by Electronic Disk)

ENCLOSURE 2
Certificate of Service for TMI-1
Technical Specification Change Request 272

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF	**DOCKET NO. 50-289**
GPU NUCLEAR INC.	**LICENSE NO. DPR-50**

CERTIFICATE OF SERVICE

This is to certify that a copy of License Amendment Request No. 272 to Appendix A of the Operating License for Three Mile Island Nuclear Station Unit 1, has, on the date given below, been filed with executives of Londonderry Township, Dauphin County, Pennsylvania; Dauphin County, Pennsylvania; and the Pennsylvania Department of Environmental Resources, Bureau of Radiation Protection, by deposition in the United Sates mail, addressed as follows:

Mr. Darryl LeHew, Chairman
Board of Supervisors of
Londonderry Township
R.D. #1, Geyers Church Road
Middletown, PA 17057

Ms. Sally Klein, Chairman
Board of County Commissioners of
Dauphin County
Dauphin County Courthouse
Front & Market Streets
Harrisburg, PA 17101

Director, Bureau of Radiation Protection
PA Dept. of Environmental Resources
Rachael Carson State Office Building
PO Box 8469
Harrisburg, PA 17105-8469
Att: Mr. Stan Maingi

GPU NUCLEAR INC.

BY: James W. Langenbach
Vice President and Director, TMI

DATE: 10/19/98