GPU NUCLEAR

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October 15, 1998
1920-98-20526

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Gentlemen:

Subject: Three Mile Island Nuclear Station, Unit 1 (TMI-1)
Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 276
Revised Accident Analysis Atmospheric Dispersion Factors

In accordance with 10 CFR 50.4(b)(1) and 10 CFR 50.59(c), please find enclosed License Amendment Request No. 276.

This license amendment request proposes a revision to the TMI-1 Updated Final Safety Analysis Report (UFSAR) Chapter 14 postulated accident analysis radiological dose consequences resulting from application of revised atmospheric dispersion factors (X/Q) at the Technical Specification Section 5.1.1 defined exclusion area boundary and low population zone. The revised atmospheric dispersion factors utilize recent meteorological data and a larger population for statistical determinations and therefore provides a higher confidence that the values are representative for accident analysis. The revised values are higher than those previously analyzed. However, the resulting postulated dose consequences at the exclusion area boundary and low population zone remain well within the guideline of 10 CFR Part 100 and within Standard Review Plan guidance.

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Using the standards in 10 CFR 50.92, GPU Nuclear has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Also enclosed is the Certification of Service for this request certifying service to the chief executives of the township and county in which the facility is located, as well as the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection.

Sincerely,

James W. Langenbach

James W. Langenbach
Vice President and Director, TMI

/DJD

Encl. (1) TMI-1 License Amendment Request No. 276 Safety Evaluation, No Significant Hazards Consideration, UFSAR Revised Pages

(2) Certificate of Service for TMI-1 License Amendment Request No. 276

cc: Administrator, Region I
TMI Senior Resident Inspector
TMI-1 Senior Project Manager
File No. 98161

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER & LIGHT COMPANY
AND
PENNSYLVANIA ELECTRIC COMPANY
THREE MILE ISLAND NUCLEAR STATION, UNIT 1

Operating License No. DPR-50
Docket No. 50-289
License Amendment Request No. 276

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF DAUPHIN)

This License Amendment Request is submitted in support of Licensee's request to change the Updated Final Safety Analysis Report (UFSAR) for Three Mile Island Nuclear Station, Unit 1. As a part of this request, proposed replacement pages for the UFSAR are also included.

GPU NUCLEAR INC.

BY: James W Langenbach
Vice President and Director, TMI

Sworn and Subscribed to before me
this 15th day of October, 1998

Suzanne C. Miklosik
Notary Public

Notarial Seal
Suzanne C. Miklosik, Notary Public
Londonderry Twp., Dauphin County
My Commission Expires Nov. 22, 1999
Member, Pennsylvania Association of Notaries

Enclosure 1

TMI-1 License Amendment Request 276 Safety Evaluation
No Significant Hazards Consideration and
Proposed UFSAR Revised Pages

I. License Amendment Request

GPU Nuclear requests that TMI-1 Updated Final Safety Analysis Report (UFSAR) Chapter 14 be revised to identify the postulated radiological dose consequences resulting from application of updated atmospheric dispersion factors (X/Q).

Revised TMI-1 UFSAR Chapter 14 pages indicating the proposed changes are attached. TMI-1 UFSAR Section 2.5.4 will be updated to include the revised X/Q methodology as described below, upon approval.

II. Reason for Change

Atmospheric dispersion coefficients for the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) have been recalculated using Regulatory Guide 1.145 methodology to ensure consistency with the boundary of the TMI-1 exclusion area as defined in TMI-1 Technical Specification Section 5.1.1, and to incorporate recent meteorological data from 1992, 1993, 1995, and 1996 with data recovery above 90% (Reference Pickard, Lowe & Garrick Report "Accident X/Q Values for TMI-1, dated July 23, 1998").

III. Safety Evaluation Justifying Change

The revised atmospheric dispersion coefficients utilized for the UFSAR Chapter 14 accident analyses 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 analyses assumed a variable EAB boundary.

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 EAB distances in each of the sixteen direction sectors represent the minimum distance from the edge of the reactor building to the EAB as defined in TMI-1 Technical Specification Figure 5-1. Table 1 provides a list of distances to the EAB in each direction sector. 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. A summary of the results for both the EAB and LPZ are given in Table 2.

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 I 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 data base 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 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 sixteen direction sectors for the 4-year period. An envelope was constructed around all sixteen 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 1. 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 2 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 2 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 3. As shown in Figure 4, 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 4. 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 5. Results are summarized in Table 2.

An evaluation of the accidents analyzed in Chapter 14 of the TMI-1 FSAR was performed to determine the impact of the revised EAB and LPZ X/Q on the offsite radiation dose consequences (Reference GPU Nuclear Calculation C-1101-900-E000-069, Rev. 2). These accidents were not re-analyzed, but evaluated, since the total activity released during each accident, as well as the time distribution over which the activity is released are identical to those described in the FSAR. Since dose is directly proportional to X/Q, the maximum offsite radiation dose at the EAB will increase by the ratio of the "revised" X/Q to the FSAR X/Q. This ratio is calculated below.

$$\frac{\text{"revised" X/Q}}{\text{FSAR X/Q}} = \frac{8.0E-4}{6.8E-4} = 1.18$$

Two Hour Exclusion Area Boundary Doses (Rem)

Accident/Transient	Revised Dose Thyroid	Revised Dose WB	FSAR Dose Thyroid	FSAR Dose WB
Maximum Hypothetical Accident (MHA)*	189	7.6	189	7.6
Main Steam Line Break (MSLB)*	28	<1	28	<1
OTSG Tube Rupture	1.26	0.36	1.07	0.31
Fuel Handling Accident in FHB	1.51	0.31	1.28	0.26
Fuel Handling Accident in RB	73.3	1.54	62.3	1.31
Rod Ejection Accident	5.2	0.007	4.43	0.006
Waste Gas Decay Tank Rupture	7.2	2.11	6.13	1.79
Spent Fuel Cask Drop Accident*	3.38	0.051	3.38	0.051
Loss of Load Transient	0.026	N/E	0.022	N/E
Station Blackout	0.49	N/E	0.42	N/E

10 CFR 100 Limits are 300 Rem to the Thyroid and 25 Rem Whole Body

N/E – Not Evaluated

*Doses for these accidents are maintained at the FSAR values because the FSAR values are bounding since they are based on a higher X/Q than the revised value of 8.0E-4sec/m^3.

Since the total activity released and the time distribution of release are unchanged, the revised LPZ dose consequences are conservatively established by determining the ratio of the revised X/Q to the X/Q used in the FSAR analysis for each period and multiplying the existing FSAR doses by the highest of the ratios as shown below (9.75 for the 8-24 hour period).

Interval	FSAR X/Q_1	Revised X/Q_2	Ratio $X/Q_2/X/Q_1$
0-2 hrs.	2.00E-05	1.40E-04	7.00
2-8 hrs.	2.00E-05	6.00E-05	3.00
8-24 hrs.	4.00E-06	3.90E-05	9.75
1-4 days	2.70E-06	1.60E-05	5.93
4-30 days	1.30E-06	4.00E-06	3.08

The only LPZ doses calculated in the FSAR are for the Maximum Hypothetical Accident, the Main Steam Line Break, and the Rod Ejection Accident, since these are the only accidents where releases are assumed to occur beyond the initial two hours of the accident. The above LPZ X/Q ratios are only applicable to the MHA and Rod Ejection Accident. For the MSLB, the existing FSAR analysis LPZ X/Q was larger than the revised X/Q and so the LPZ doses are conservatively maintained at their current value. Recalculation produces the following results.

30 Day LPZ Doses (Rem)

Accident/Transient	Revised Dose Thyroid	Revised Dose WB	FSAR Dose Thyroid	FSAR Dose WB
Maximum Hypothetical Accident (MHA)	85.8	2.05	8.8	0.21
Rod Ejection Accident	9.72	0.009	0.997	0.0009
Main Steam Line Break	5.3	<1	5.3	<1

10 CFR 100 Limits are 300 Rem to the Thyroid and 25 Rem Whole Body

The revised X/Q values have increased due to the evaluation of a larger and more recent data base of meteorological data than that used for originally licensing the plant. This larger data base provides a more accurate representation of X/Q values than previously utilized. The increased X/Q values result in increases in the EAB and LPZ dose consequences for the postulated radiological accidents analyzed in TMI-1 FSAR Chapter 14. However, the revised dose consequences remain well below 10 CFR 100 guidelines.

IV. Environmental Consideration

GPU Nuclear has determined that this change to the TMI-1 FSAR Chapter 14 accident analysis atmospheric dispersion factors and radiological dose consequences involves no significant change in the amount or type of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The revised radiological consequences of the effected TMI-1 FSAR Chapter 14 accidents are below 10 CFR 100 limits for the EAB and LPZ. As such, operation of TMI-1 in accordance with the proposed change does not involve an unreviewed environmental safety question.

V. No Significant Hazards Consideration

GPU Nuclear has determined that this License Amendment Request poses no significant hazards as defined by 10 CFR 50.92.

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 effect on structures, systems or components. More extensive and recent meteorological data have been utilized for atmospheric dispersion factor (X/Q) determination for both EAB and LPZ. An evaluation of the design basis accidents with revised EAB and LPZ X/Q values results in increases in UFSAR Chapter 14 EAB and LPZ dose consequences which remain well within the guidelines of 10 CFR Part 100. 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. The proposed change revises the atmospheric dispersion factors for EAB and LPZ used in the existing UFSAR Chapter 14 accident analyses, based on more extensive meteorological data. These changes only effect the postulated dose consequences of currently analyzed accidents. 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. The proposed revisions to the EAB and LPZ X/Q values are based on recent more extensive meteorological data and Regulatory Guide 1.145 methods. The increased X/Q values provide a more accurate assessment of meteorological conditions which result in postulated dose consequences which remain well within the guidelines of 10 CFR Part 100. Therefore, this activity does not reduce the margin of safety.

VI. Implementation

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

Attachment I

1992, 1993, 1995, 1996 Meteorological Data
(via magnetic media: electronic diskette)

TABLE 1

DISTANCES TO EAB IN EACH DIRECTION SECTOR	
Direction Sector	**Distance (m)**
N	588
NNE	591
NE	591
ENE	594
E	594
ESE	594
SE	686
SSE	686
S	725
SSW	686
SW	686
WSW	591
W	588
WNW	588
NW	588
NNW	588

TABLE 2

Results of 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 4

**Values in the NE direction would be lower

Figure 1

X/Q Values Based on 4 Years of TMI Data (92, 93, 95, 96)
(Direction Is From the Indicated Sector)

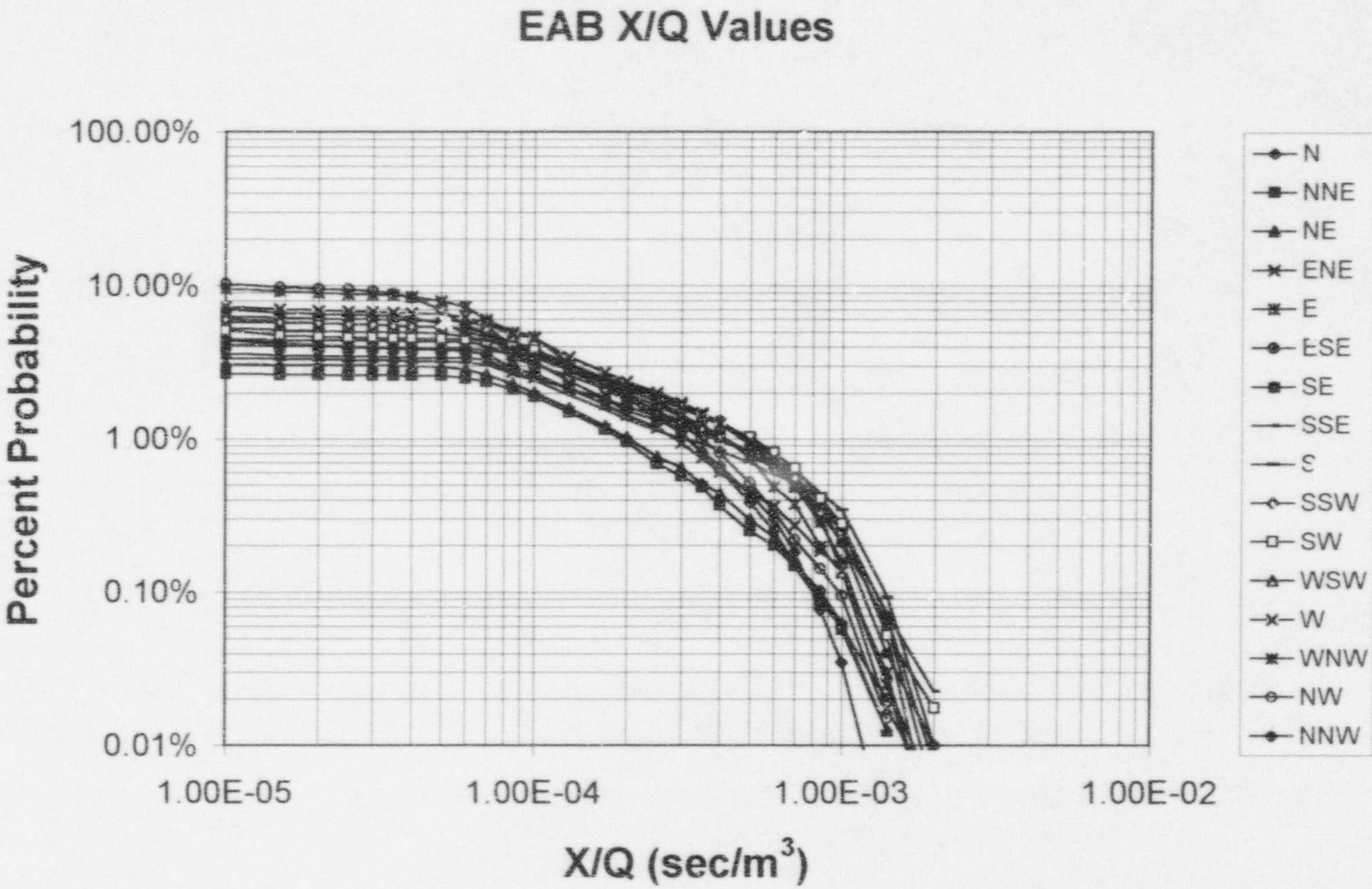
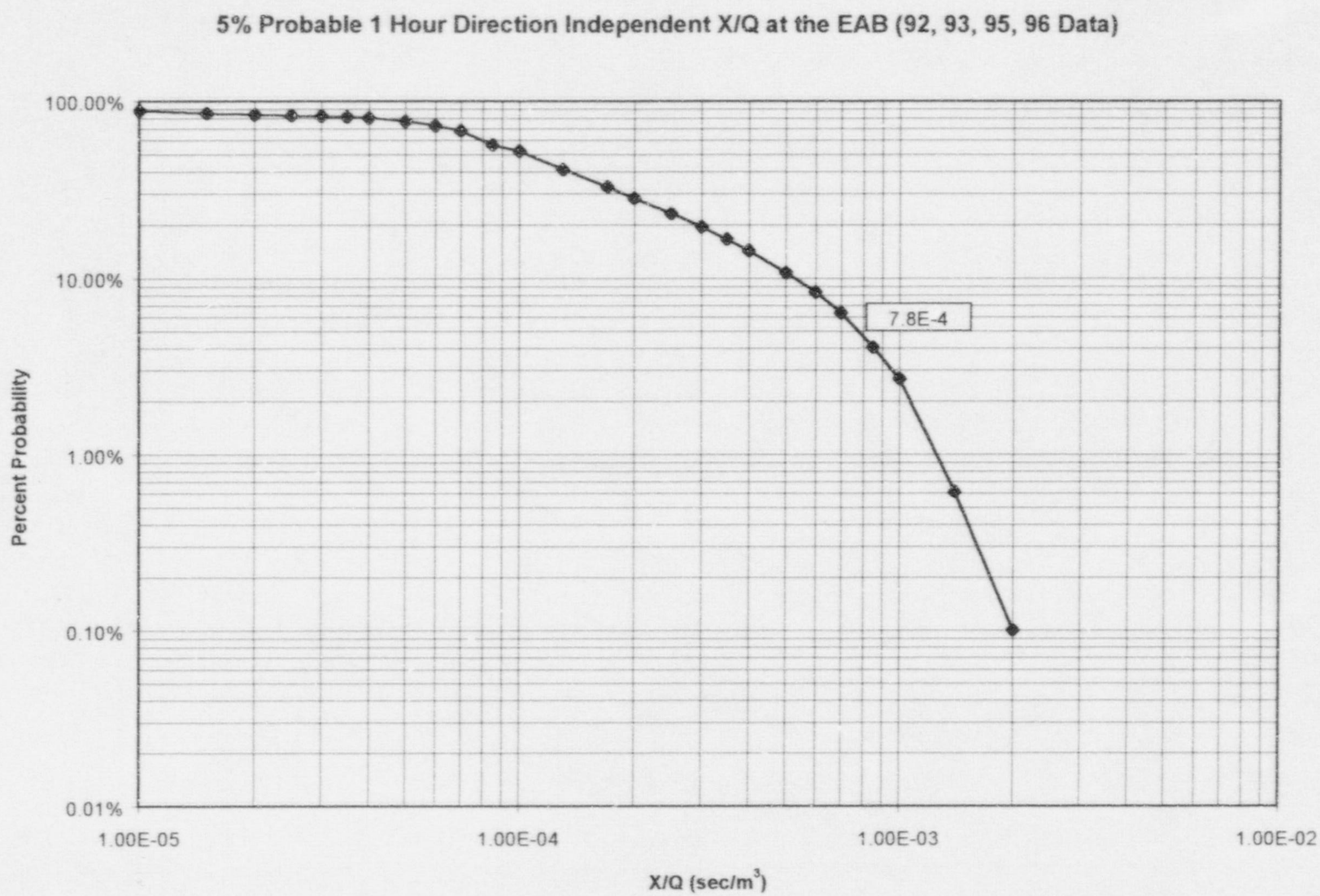


Figure 2



5% Probable 1 Hour Direction Independent X/Q at the EAB (92, 93, 95, 96 Data)
100.00%
10.00%
1.00%
0.10%
0.01%
Percent Probability
7.8E-4
1.00E-05
1.00E-04
1.00E-03
1.00E-02
X/Q (sec/m3)

Figure 3

X/Q Values Based on 4 Years of TMI Data (92, 93, 95, 96)
(Direction Is From the Indicated Sector)

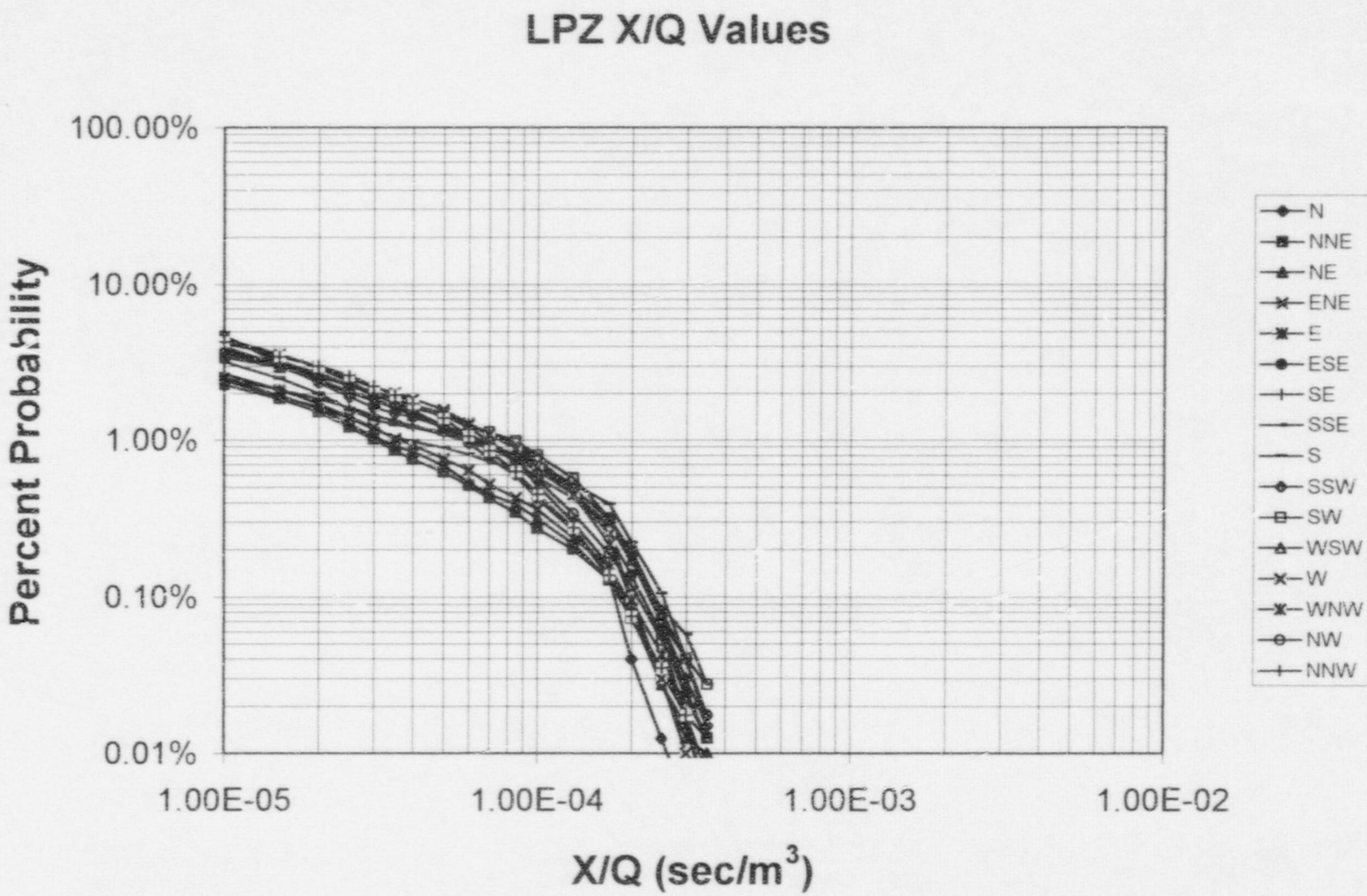


Figure 4

LPZ X ; Values Based on 4 Years of TMI Data (92, 93, 95, 96)

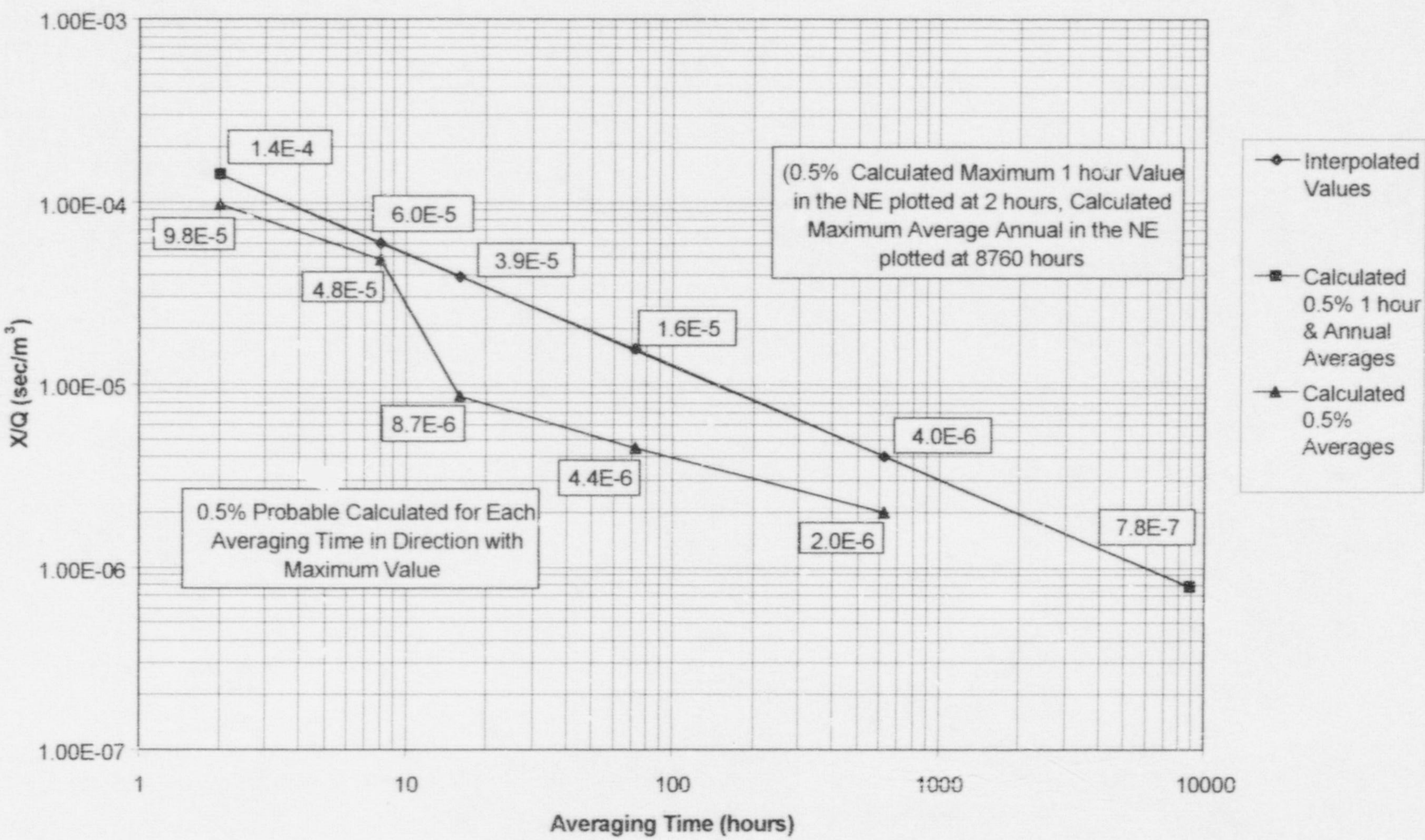
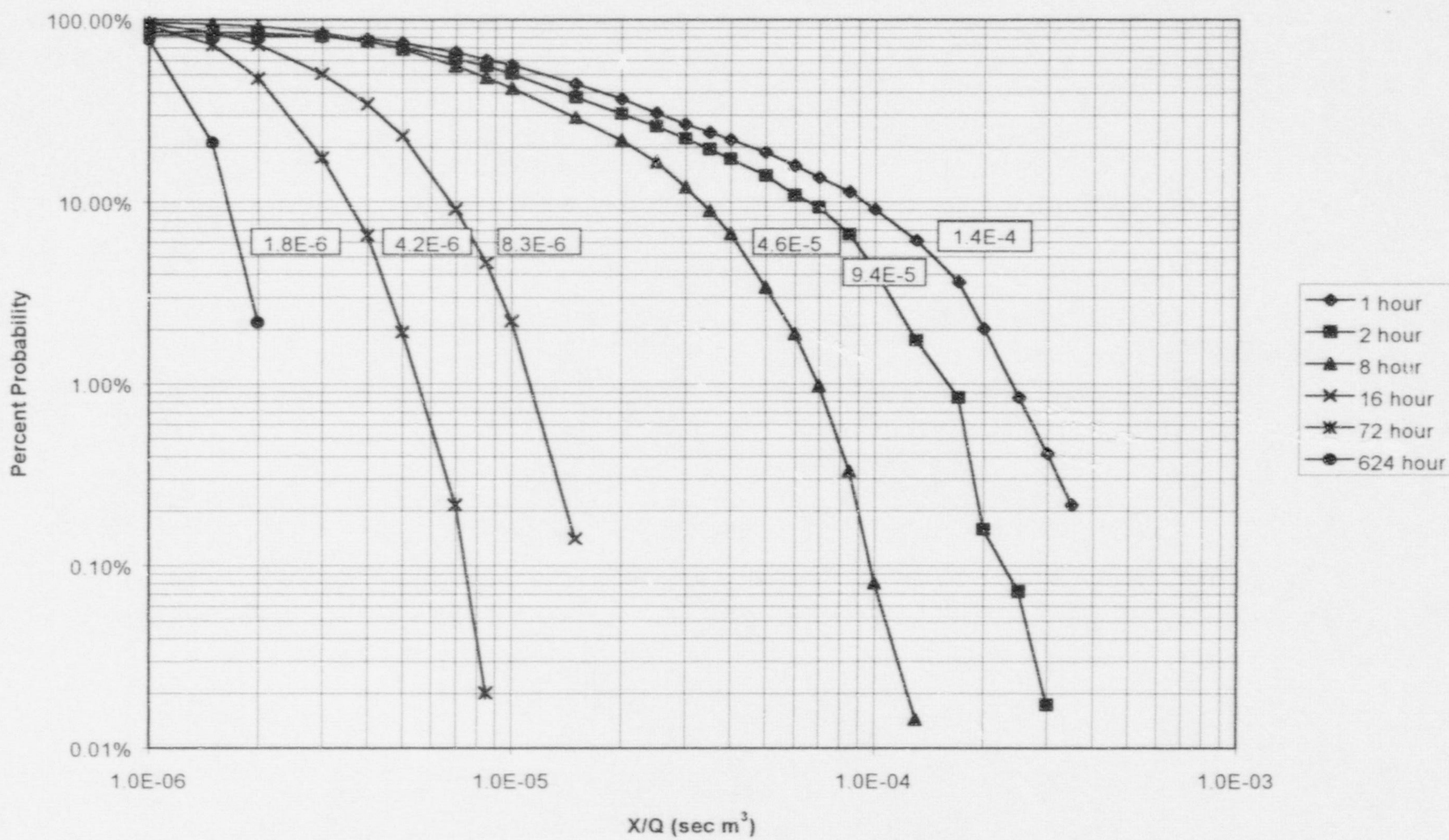


Figure 5

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



Proposed UFSAR Revised Pages

14.1-1
14.1-57
14.1-58
14.1-64
14.2-2
14.2-5
14.2-16
14.2-45
14.2-53
14.2-55
14.2-62
14.2-76
14.3-5a
14C-7

14.0 SAFETY ANALYSIS

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1 ABNORMALITIES

In previous Chapters of this report, both normal and abnormal operations of the various systems and components have been discussed. This Chapter summarizes and further explores abnormalities that either are inherently terminated or require the normal protection systems to operate to maintain integrity of the fuel and/or the Reactor Coolant System. Most of these abnormalities have been evaluated for a core power greater than or equal to 2535 MWt. Table 14.1-1 summarizes the potential abnormalities studied.

For TMI-1 Cycle 7 reload design, the rated power was upgraded from 2535 MWt to 2568 MWt. Each accident analysis in this chapter has been examined with respect to the power upgrade and corresponding accident parameter changes. The safety evaluation concluded that the power upgrade does not present any adverse safety impact and that the previously accepted design basis (safety analysis parameters) used in the FSAR bounds the power upgrade parameters (Reference 62).

The radiological source term data were reevaluated to reflect: (1) power upgrade to 2568 and (2) slight increase radioactivities due to increased fission yields of Pu-239. Furthermore, the Cycle 7 core inventory was deliberately increased by applying a conservatism factor of 1.10 for the purpose of enveloping future cycle variations. The updated source term data are presented in Table 14.2-4.

The fission product dispersion factor (X/Q-value) at the exclusion boundary is 8.0×10^{-4} sec/M^3 (Reference 82) unless specified otherwise in the following accident analysis discussion.

The radiological dose consequence based on the updated source term data are within the 10CFR100 dose acceptance criteria and are incorporated in each subsection below.

Additional documentation and references to specific calculational methods and criteria are identified in Reference 69.

TABLE 14.1-14
(Sheet 1 of 1)

LOSS OF LOAD TRANSIENT PARAMETERS AND RESULTS

Initial RCS flow, gpm	352,000 (100% flow)
Initial Core Power, MWt	2568
Moderator Coefficient at BOL (delta-k/k)/°F	+0.00
Doppler Coefficient at BOL	-1.17×10^{-5}
Steam relieved to the atmosphere lbm	205,000
Atmospheric dispersion coefficient at exclusion distance sec/m^3	8.0×10^{-4}
Iodine released during relief (in Iodine-131 dose equivalent curies)	6.26×10^{-2}
Total integrated thyroid dose at exclusion distance rem	2.60×10^{-2}

TABLE 14.1-15
(Sheet 1 of 1)

LOSS OF ALL A-C POWER EVENT (STATION BLACKOUT) PARAMETERS AND RESULTS

Steam relieved to atmosphere, lbm	203,900
Atmospheric dispersion coefficient at exclusion distance, sec/m^3	8.0×10^{-4}
Steam generator isolation time, min	55
Total integrated thyroid dose at exclusion distance, Rem	4.90×10^{-1}
Initial RCS flow, gpm	352,000 (100% flow)
Initial Core Power, MWt	2568
Moderator Coefficient at BOL (delta-k/k)/°F	+0.00
Doppler Coefficient at BOL	-1.17×10^{-5}

TABLE 14.1-21
(Sheet 1 of 1)

SUMMARY OF STEAM GENERATOR TUBE FAILURE ANALYSIS

Low pressure trip occurs at min	8
Total depressurization time of Reactor Coolant System min	34
Reactor coolant leakage during depressurization ft^3	1977
Activity Released to Atmosphere	
Noble gases curies	32,400
Iodine I-131 dose equivalent curies	3.06
Total Integrated Dose at Exclusion Distance	
Thyroid Rem	1.26
Whole body Rem	0.36

The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the Fuel Handling Building atmosphere. Normally, the spent fuel assembly rests within the spent fuel storage rack, where it is covered with a minimum of 23 feet of water. During spent fuel handling, the minimum water depth above the top of the active fuel is 6.5 feet. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. In experiments whereby air-steam mixtures were bubbled through a water pond, Reference 14 demonstrated decontamination factors of about 1000 for iodine. Similar results for iodine were demonstrated in References 15 and 16. Based on these References, 99 percent of the iodine released from the fuel assembly is assumed to remain in the water, which is consistent with the requirements given in Regulatory Guide 1.25. The Fuel Handling Building is ventilated and discharges through 90 percent efficient charcoal filters to the unit vent.

The activity is assumed to be released as a puff from the unit vent. Atmospheric dilution is calculated using the 2 hour dispersion factor at the exclusion boundary of 8.0×10^{-4} sec/m^3, which is discussed in Reference 82. The total integrated dose at the exclusion distance to the whole body and to the thyroid can be seen in Table 14.2-3.

A Fuel Handling Building environmental barrier is included to limit potential leakage paths and isolate Unit 1 refueling floor from the Unit 1 Auxiliary Building. Ventilation systems are designed to prevent leakage path. See Section 9.8.2.2.

2) Fuel Handling Accident Occurring in the Reactor Building

An evaluation of the postulated Fuel Handling Accident Inside Containment (FHAIC) at TMI-1 has been performed according to NRC request. No credit was taken for Reactor Building isolation, no changes to facility equipment or Technical Specifications have been considered. The proposed operation and surveillance Technical Specifications on the Reactor Building Purge Exhaust System (RBPES) assure a credit of 70 percent for total radioiodine removal efficiency.

In addition, radiation monitors RM-G6 and RM-G7 monitor and alarm any excessive radiation in the vicinity of the refueling water surface. Also, Technical Specification 3.8.5, which requires that direct communications between the Control Room and the refueling personnel in the Reactor Building, is in effect whenever changes in core geometry are taking place. Therefore, by radiation monitoring and Technical Specification implementation assurance is provided that in the event of a fuel handling accident in the Reactor Building, the Control Room operators would have sufficient information to initiate isolation of the Reactor Building.

j) Atmospheric diffusion is calculated using a 0-2 hr dispersion factor at the exclusion boundary of 8.0×10^{-4} sec/m^3. This value is discussed in Reference 82

The analysis of the radiological consequences of a FHAIC has been performed taking credit for Reactor Building Purge Exhaust System (RBPES). The two-hour dose results at the exclusion boundary are given in Table 14.2-5.

Because there is a chance that more than one spent fuel assembly may be damaged during refueling, the probability and consequences of dropping a spent fuel assembly in the core and damaging more fuel pins than the equivalent of one assembly was also discussed (Reference 17). The conclusion is that the doses for failure of two assemblies would not be greater than the exposure guidelines of 10CFR100 and no additional restrictions on fuel handling operations and plant operation procedures are need.

14.2.2.2 Rod Ejection Accident

a. Identification of Accident

Reactivity excursions initiated by uncontrolled rod withdrawal (Section 14.1) were shown to be safely terminated without damage to the reactor core or Reactor Coolant System integrity. For reactivity to be added to the core at a more rapid rate, physical failure of a pressure barrier component in the control rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by Reactor Protection System trips.

generator pressure was low enough to allow critical flow during the accident. This yields a maximum leak flow rate. It is also conservatively assumed that the friction factor associated with the leak flow area was equal to 1.0, and that the discharge coefficient was equal to 1.0, and that the discharge coefficient was equal to 1.0. Using these assumptions, it is calculated that 5 gallons of reactor coolant leak to the secondary system during the time the Reactor Coolant System is depressurizing. The activity associated with this 5 gallons is released to the atmosphere from the condenser. A gas to liquid partition factor of 10-4 assumed for the iodine in the condenser, (References 10 and 11), but the noble gases are assumed to be released directly to the atmosphere. The activity released directly to the atmosphere is shown in Table 14.2-11.

All reactor coolant that is not released to the secondary system is released to the Reactor Building. Fifty percent of the iodine released to the Reactor Building is assumed to plate out. The activity released to the Reactor Building is shown in Table 14.2-11.

Fission product activities for this accident are calculated using the methods discussed in Chapter 11. Doses resulting from this accident were evaluated using the environmental models and dose rate calculational methods discussed in the section on the loss of coolant accident. Table 14.2-11 shows the resulting thyroid and whole body doses for a 2 hr exposure at the exclusion distance and for a 30 day exposure at the low population distance. In addition to the total thyroid and whole body doses, the table includes the dose contribution due to the activity released to the atmosphere via the secondary system and that released via Reactor Building leakage. Activity released due to normal operation within the Technical Specification limits was not considered in this accident analysis and is considered to be negligible. The doses resulting from the accident are well below the guideline values of 10CFR100.

The dose results in Table 14.2-11 were based on a 0.65 percent delta-k/k rod ejection accident with a nominal moderator temperature coefficient. As shown on Figure 14.2-7, the rod ejection accident results are not strongly dependent on the value of the moderator coefficient. Even if the Technical Specification maximum moderator coefficient of 0.5×10^{-4} (delta-k/k)°F were used in the 0.65 percent delta-k/k rod ejection accident evaluation, the 5.2 rem total integrated 2 hr thyroid dose at the exclusion distance would increase by only about 1.4 rem.

A tank is assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1 percent defective fuel. The reactor coolant passes through purification demineralizers which remove 99 percent of the iodine. The coolant is then degassed an additional 99 percent according to the liquid/gas partitioning for iodine. The resulting waste gas inventory is 0.01 percent of the iodine and all of the noble gas activities associated with one reactor coolant volume. All of this activity is assumed to be released to the Auxiliary Building. Charcoal filters with 90 percent efficiency remove more iodine as the radioactivity releases into the environment. The resulting leakage to the environment is 10 percent of the water gas tank iodine, or 0.001 percent of the reactor coolant iodine, and all of the reactor coolant noble gases.

The gaseous activity in the tank is listed in Table 14.2-21.

The Auxiliary Building is ventilated and discharges to the unit vent. The activity is assumed to be instantaneously released as a puff. No radioactive decay is accounted for. Additionally, no removal mechanisms for noble gases are assumed. The total integrated doses at the exclusion distance are:

2 Hour Doses at Exclusion Distance

Thyroid Dose	7.2 Rem
Whole Body Dose	2.11 Rem

These doses are well below the limits of the 10CFR100 guideline.

14.2.2.7 Loss Of Feedwater Accident

a. Identification

A loss of feedwater may result from abnormal closure of the feedwater isolation valves, control valve failure, or pump failure. The loss of feedwater flow results in a loss of heat sink, primary system heatup, increased pressurizer level and pressure, and reactor trip on high RCS pressure.

Acceptance Criteria

For the transient analyses, the acceptance criteria chosen were Prevention of Pressurizer Fill and, Prevention of Saturated Condition in the RC Hot Leg. The general criteria are as follows:

1) Core thermal power shall not exceed 112 percent of rated power.

TABLE 14.2-3
(Sheet 1 of 1)

FUEL HANDLING ACCIDENT PARAMETERS AND RESULTS
(IN THE FUEL HANDLING BUILDING)

Power level (MWt)	2568
Damaged fuel rods in the assembly	56
Spent fuel pool water Decontamination factor (Iodine)	100
Charcoal filter efficiency (Iodine), %	90
Atmospheric dilution factor sec/m^3	8 $\times 10^{-4}$
Total activity release time (hour)	2
Total Integrated Dose at Exclusion Distance	
Thyroid Rem	1.51
Whole Body Rem	0.31

TABLE 14.2-5
(Sheet 1 of 1)

RADIOACTIVE RELEASE FOR(*)
THE POSTULATED FUEL HANDLING
ACCIDENT AND DOSE RESULTS
(In the Reactor Building)

Isotope	Activity Released (curies)(**)
Kr-83m	9.77×10^3
85	2.00×10^4
87	4.22×10^4
88	5.91×10^4
Xe-131m	4.68×10^2
133m	3.24×10^3
133	1.34×10^5
135	2.30×10^4
I-131	7.85×10^2
132	9.15×10^2
133	1.35×10^3
134	1.70×10^3
135	1.34×10^3

TWO HOUR DOSE RESULTS	DOSE (REM)
Thyroid	73.3
Whole Body	1.54

(*) Curies = (Core Inventory in Table 14.2-4) x $\frac{1.7}{177}$

(**) Regulatory Guide 1.25 Release Fraction and Refueling Pool Iodine Decon Factor of 100 are applied.

TABLE 14.2-11
(Sheet 2 of 2)

ENVIRONMENTAL EFFECTS OF ROD EJECTION ACCIDENT

	Total Dose (rem)
Two Hour Dose at Exclusion Distance	
Thyroid	5.2
Whole body	0.007
30-Day Dose at Low Population Distance	
Thyroid	9.72
Whole body	0.009

TABLE 14.2-20
(Sheet 1 of 1)

ENVIRONMENTAL DOSES RESULTING FROM MHA

2-Hour Dose at Exclusion Distance (Appendix 14C), rem*	
Thyroid	189
Whole body	7.6
30-Day Dose at Low Population Distance, rem*	
Thyroid	85.8
Whole body	2.05
Engineered Safeguards Leakage	
Iodine Concentration in Liquid, I-131 dose equiv Ci/ml	0.033
Liquid Leakage, ml/hr	2,165
Leakage That Flashes, ml/hr	90
Thyroid Dose at Exclusion Distance, Rem	0.037
Reactor Building Design Leak Rate	0.1%/day
Charcoal Filter Efficiency	90%
AEC Staff Model (Reference 55)	
2-hour meteorology, sec/m^3	1.2×10^{-3}
2-hour iodine dose reduction factor	5.58
2-hour thyroid dose at exclusion distance, rem	269

* The result of the AEC Staff Model are based on the original licensed power rating of 2535 MWt. The current values of 2 hr and 30 day MHA doses on this table are based on licensed power upgrade to 2568 MWt.

73. BWNT Document 86-1232708-00, "RVVV Liquid Spillover Analysis," September 1994.

74. GPUN Document C-1101-211-5450-008, "RCS Boron Concentration with Credit for Vent Valve Liquid Flow," September 19, 1994.

75. BWFC Letter, RC-94-786, "TMI-1 Cycle 11 Steam Line Break Evaluation," 12/1/94.

76. GPUN SE 135400-022, Rev. 2., "TMI Cycle 11 Reload Design/Redesign," 11/10/95.

77. Framatome Technologies Document 43-10222P-00, "TMI-1 2772 MWt ECCS Analysis with RELAP5/MOD2-B&W", dated March 1997.

78. Framatome Technologies Document 43-10192P-00, "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", dated February 1994.

79. J. A. Klingenfus, et al., "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164, Revision 3, B&W Nuclear Technologies, Lynchburg, Virginia, October 1992.

80. C. K. Nithianandan, "REFLOD3B - Model for Multinode Core Reflooding Analysis, BAW-10171, Revision 3, B&W Nuclear Technologies, Lynchburg, Virginia, September 1989.

81. N. H. Shah, et al., "BEACH – A Computer Program for Reflood Heat Transfer During LOCA," BAW-10166, Revision 4, B&W Nuclear Technologies, Lynchburg, Virginia, October 1992.

82. Pickard, Lowe & Garrick Report "Accident X/Q Values for TMI-1", dated July 23, 1998.

TABLE 14C-1
(Sheet 1 of 1)
ASSUMPTIONS USED FOR DOSE CALCULATIONS

Time period, T	2 hours
Distance to exclusion boundary	610m
Average atmospheric diffusion factor, X/Q	8.3×10^{-4} sec/m^3
Breathing rate, BR	3.47×10^{-4} m^3/sec
Power level, P_O	2535 MW (Note 2)
Containment leakage rate, L	0.12%/day (Note 3)
Elemental iodine initial release fraction, α_E	0.23875 (Note 4)
Organic iodine initial release fraction, α_o	0.0100
Particulate iodine initial release fraction, α_p	0.0125
Spray removal coefficients, λi	see Table 14B-3
Decontamination factors, DF_i	see Table 14B-3
Total containment free volume, V_C	2.16×10^6 ft^3
Containment Building sprayed volume, V_s	1.45×10^6 ft^3
Containment Building unsprayed volume, V_u	0.71×10^6 ft^3
Mixing Flow Between Sprayed and Unsprayed Volumes, F_A	
One fan operating (minimum safety)	54,000 ft^3/min
Three fans operating	162,000 ft^3/min

[1] Deleted
[2] The power was upgraded from 2535 MWt to 2568 MW_t
[3] Technical Specifications restrict the containment leakage rate to 0.1%/day
[4] Credit is taken for instantaneous plateout

ENCLOSURE 2
Certificate of Service for TMI-1
License Amendment Request No. 276

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF
GPU NUCLEAR INC.

DOCKET NO. 50-289
LICENSE NO. DPR-50

CERTIFICATE OF SERVICE

This is to certify that a copy of License Amendment Request No. 276 to the Facility 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: 10/15/98