Safety Evaluation Report

NUREG-0107
SUPP 1

U. S. Nuclear Regulatory Commission

related to operation of

Office of Nuclear Reactor Regulation

Three Mile Island Nuclear Station, Unit 2

Docket No. 50-320

Metropolitan Edison Company
Jersey Central Power & Light Company
Pennsylvania Electric Company

Supplement No. 1

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SUPPLEMENT NO. 1

TO THE

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION
UNITED STATES NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY

THREE MILE ISLAND NUCLEAR STATION
UNIT 2

DOCKET NUMBER 50-320

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1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

Since publication of the Safety Evaluation Report in September 1976, we have received and reviewed several amendments to the Final Safety Analysis Report, held a number of meetings with the applicant, and met with the Advisory Committee on Reactor Safeguards. These events are identified in the Chronology, Appendix A to this supplement. As a result of these actions, many of the issues identified as outstanding review items in Section 1.7 of the Safety Evaluation Report have been resolved.

This supplement provides (1) our evaluation of additional information received from the applicant since issuance of the Safety Evaluation Report, (2) the status of previously identified outstanding review items, including those now resolved and those still under review, (3) a discussion of items identified by the Advisory Committee on Reactor Safeguards in its report of October 22, 1976, and (4) additional or revised information for certain sections of the Safety Evaluation Report as appropriate.

Sections of this supplement carry the same numbers as those of the Safety Evaluation Report which they supplement or modify, and except where specifically noted, do not replace sections of the Safety Evaluation Report.

1.7 Summary of Outstanding Review Items

Many items previously identified as outstanding have been resolved since publication of the Safety Evaluation Report. In addition, for some issues, new information has been received and new references added. The current status of each item is identified below. For issues identified as resolved, additional discussion is presented in the referenced sections of this supplement.

-- Redesign and/or reanalysis of the containment spray chemical additive system (Sections 6.2.3, 15.3.1).

   This issue has been resolved.

-- Seismic qualification of safety-related display instrumentation (Section 7.5).

   This issue has been resolved.

-- Completion of revised steam line break analysis (Sections 10.3, 10.5, 7.3.3, 7.4.1, 6.2.1, 15.2.2).

The applicant submitted a revised steam line break analysis in Amendment 50, which we have reviewed and found unacceptable. We have identified to the applicant our requirements for additional analyses and/or system modifications.

-- Reactor cavity pressure response analysis confirmation (Section 6.2.1).

The applicant has submitted analyses on a further revised design. Our review of these analyses has not yet been completed.

-- Reactor building sump test results (Sections 6.2.2, 6.3.4).

Although the test results have been available to us, the applicant has not yet submitted the sump test report.

-- Installation of flow measurement devices to assure adequate flow to limit boron concentration (Section 6.3.3).

The applicant has submitted a description of the equipment provided to accomplish this flow measurement. Our review of this information is not yet complete.

-- Schematics verifying design implementation of:

a. Automatic switchover from injection mode to recirculation mode (Section 7.6.1).

b. Bypass diesel generator trip devices (Section 8.3.1).

c. Testability of engineered safety feature undervoltage relays (Section 8.3.1).

This issue has been resolved.

-- Qualification as isolation devices of output relays in the safety features actuation system (Section 8.3.1).

We have evaluated information provided by the applicant, and will transmit clarification of our requirements for testing these relays to the applicant.

-- Data on iodine removal capability of two-inch charcoal bed depths (Section 11.3).

This issue has been resolved.

-- Descriptions of test programs and methods for:

a. Reactor protection system and engineered safety feature actuation system response time testing (Sections 7.6.4, 14.0).

b. Control rod drop time testing to full insertion (Section 14.0).

We have reviewed information provided by the applicant and have identified additional information required.

-- Evaluation of data on environmental qualification of balance-of-plant Class IE electrical equipment (Section 7.8.2).

We have transmitted our requirements for additional information to the applicant.

-- Appendix I review (Section 11.1).

This issue has been resolved.

-- Review of feed line break analysis (Section 15.2.2).

We have reviewed the applicant's analysis and, in conjunction with our review of the steam line break analysis, have identified requirements for additional analyses and/or system modifications.

The following items were identified in the Safety Evaluation Report as generic in nature.

- Evaluation of reactor vessel supports (Section 3.9.3).

  This item is being pursued and monitored as part of our effort on generic items (see Appendix C, Item II C-5).

- Pellet cladding mechanical interaction (Section 4.2.1).

  This item has been resolved.

- Startup overpressure protection (Section 5.2.2).

  Our requirements in this area are being evaluated relative to the systems and procedures employed in this facility.

- Emergency core cooling analysis modifications (Section 6.3.3).

  Proposed revisions by Babcock & Wilcox to their generic analyses, which the applicant has stated are applicable to Three Mile Island Unit 2, were received and are under review.

- Anticipated transients without scram (Sections 7.2.2, 15.2.1).

  The discussion in Section 7.2.2 of the Safety Analysis Report notes that considering the probability of occurrence of the event in question, we conclude that limitations on operation on this account are not necessary or appropriate until such time as any facility modifications found necessary by our review of the required additional information can be implemented.

- Evaluation of electrical penetration test report (Section 7.9).

  This issue has been resolved.

2.0 SITE CHARACTERISTICS

2.3.5 Long-Term (Routine) Diffusion Estimates

In the Safety Evaluation Report, we described how we evaluated routine diffusion estimates. The following paragraph describes more completely the model and assumptions used, and includes a list of relative concentration and deposition values used in estimating the doses contributed by gaseous effluents during normal operation, including anticipated operational occurrences. It does not alter in any way the conclusions reached in Section 11.0 of the Safety Evaluation Report.

We have made reasonable estimates of average atmospheric dispersion conditions for the Three Mile Island site using our atmospheric dispersion model for long-term releases (Sagendorf and Goll, draft, 1976). This model is based upon the Straight-Line Trajectory Model described in Regulatory Guide 1.111," Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents from Light-Water-Cooled Reactors." The model assumes a mixture of elevated and ground-level releases, based on the criteria established in Regulatory Guide 1.111. Non-continuous and intermittent gaseous releases were evaluated separately from continuous releases. The calculations also include an estimate of maximum increase in calculated relative concentration and deposition due to open terrain recirculation of airflow not considered in the straight-line trajectory model. Table 2.1 lists relative concentration and deposition values used in the dose estimations which are summarized in Section 11.0.

TABLE 2.1

RELATIVE CONCENTRATION (X/Q) AND DEPOSITION (D/Q) USED IN DOSE ESTIMATE SUMMARY

Source	Receptor	Direction (From source)	Distance (mi) (From source)	X/Q Continuous Release	D/Q Continuous Release	X/Q Periodic Release (24 at 2 hrs each)	D/Q Periodic Release (24 at 2 hrs each)
Reactor Vent	Residence/ Garden	E	0.43	2.3E-06	3.8E-08	5.2E-06	1.2E-07
		W	0.37	1.9E-06	2.7E-08	4.7E-06	9.4E-08
		WNW	0.37	1.4E-06	2.2E-08	6.7E-06	1.4E-07
		N	1.00	3.0E-07	4.1E-09	9.3E-07	1.8E-08
	Cows	SE	1.2	1.1E-06	1.5E-08	2.1E-06	3.4E-08
Turbine Vent	Residence/ Garden	E	0.43	3.6E-05	1.1E-07		
		W	0.37	4.7E-05	1.3E-07		
		WNW	0.37	4.5E-05	1.1E-07		
		N	1.00	6.4E-06	1.3E-08		
	Cows	SE	1.2	5.4E-06	2.1E-08		

4.0 REACTOR

4.2 Mechanical Design

4.2.1 Fuel

In the Safety Analysis Report, we identified pellet cladding interaction as a possible fuel failure mechanism, and noted that (1) we are pursuing this problem on a generic basis, and (2) experience with Oconee 1 showed no failures related to this phenomenon. We identified a tentative schedule for our action and stated that we believe that technical specification limits on coolant activity provide adequate protection against operation with excessive failed fuel.

Our efforts have been delayed, but our review to date of the effects of pellet cladding interaction has so far not identified any safety problem. The Babcock & Wilcox fuel rod design incorporates features directed at reducing cladding strain due to pellet cladding interaction. These include pellet chamfering, prepressurization, incorporation of plenum regions at both top and bottom of the fuel rod, and thicker cladding. Based on experimental and commercial reactor data available, these design features should delay the potential for pellet cladding interaction failures to later in the fuel design lifetime. While the failure threshholds are probably lower at higher burnups, the fuel duty is also less severe. Therefore, operating restrictions are not presently warranted. If in the future our continuing program in this area should identify any safety problems, appropriate requirements would be imposed at that time. We therefore consider this matter resolved.

6.0 ENGINEERED SAFETY FEATURES

6.2.3 Containment Air Purification and Cleanup System

In the Safety Evaluation Report we noted that the proposed spray additive system was not expected to achieve its design criteria, and that we therefore required redesign of the system.

The applicant has since redesigned the sodium hydroxide portion of the system and deleted the sodium thiosulphate portion, so that sodium hydroxide is now the only spray additive.

Our evaluation of the iodine removal effectiveness of the revised system results in calculated removal rate coefficients (λ_s) of 10 inverse hours and 0.4 inverse hours for the elemental and particulate forms of iodine, respectively, in an estimated effective volume of 1.764×10^6 cubic feet, which comprises 82 percent of the total free volume of the containment. A long-term containment sump solution pH above 8.5 assures equilibrium partitioning of elemental iodine such that a decontamination factor of 100 is maintained in the containment.

We have reviewed this revised system and conclude that it can meet the design criteria with a single active failure, and is therefore acceptable. Our evaluation of the offsite doses resulting from a postulated loss-of-coolant accident is presented in Section 15.0 of this supplement.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

The Safety Evaluation Report stated that our review of the capability of the liquid and gaseous radioactive waste treatment (radwaste) systems to meet the dose objectives of Appendix I to 10 CFR Part 50 had not yet been completed.

We have since completed that review, and find that, based on our evaluation, the radwaste systems are capable of reducing releases of radioactive materials in effluents to "as low as reasonably achievable" levels which meet the individual dose design objectives set forth in Sections II.A, B and C of Appendix I. In conformance with Section II.D of Appendix I, our cost-benefit analysis concludes that there are no items of reasonably demonstrated technology that, when added to the system sequentially, can (for a favorable cost-benefit ratio) effect reductions in dose to the population reasonably expected to be within 50 miles of the reactor.

11.3 Gaseous Radioactive Waste Treatment Systems

The Safety Evaluation Report noted that the applicant had committed to furnishing data to support the use of a decontamination factor of 10 for removal of iodine by two-inch bed depth charcoal adsorbers.

In Amendment 50 to the Final Safety Analysis Report, the applicant furnished results of a test program showing that iodine removal efficiencies under appropriate temperature, humidity, and flow conditions and at normal iodine concentrations ranged between 88.8 and 97.5 percent. We find the submitted data acceptable and conclude that it supports the use of a decontamination factor of 10 for removal of iodine by two-inch bed depth charcoal filters.

We consider this matter resolved.

11.6 Conclusions

In the Safety Evaluation Report, we stated that the capability of the liquid and gaseous radwaste systems to meet the dose design objectives of Appendix I to 10 CFR Part 50 would be evaluated in a supplement to that report.

Our evaluation has been completed as described in Section 11.7 of this supplement. We conclude that there are no cost-effective augments to reduce the cumulative population dose at a favorable cost-benefit ratio.

Based on our evaluation, we conclude that the liquid and gaseous radwaste treatment systems are capable of reducing releases of radioactive materials in liquid and gaseous effluents to "as low as reasonably achievable" levels in conformance with 10 CFR Part 50.34a and meet the requirements of Appendix I to 10 CFR Part 50, and are therefore acceptable.

We consider this matter resolved.

11.7 Appendix I Evaluation

By letter of February 23, 1976, the applicant was requested to submit additional information concerning the means to be employed to keep levels of radioactive materials in effluents from Three Mile Island Unit 2 to unrestricted areas "as low as reasonably achievable" in accordance with the guidelines of Appendix I to 10 CFR Part 50, and was given the option of providing either a cost-benefit analysis or demonstrating conformance to the guidelines given in the Annex to Appendix I. The applicant's evaluation was contained in a submittal from R. C. Arnold, Vice President of Metropolitan Edison Company, to K. Kniel, Light water Reactors Branch No. 2, dated June 4, 1976. In that submittal, Metropolitan Edison Company chose to perform the cost-benefit analysis required by Section II.D of Appendix I to 10 CFR Part 50.

We have performed an independent evaluation of the applicant's methods to meet the requirements of Appendix I. Our evaluation consisted of: (1) a review of the information provided by the applicant in response to the letter of February 23, 1976; (2) a review of the applicant's radwaste treatment and effluent control systems as described in the Final Safety Analysis Report, as amended through Amendment 44 (September 1976) and in the Environmental Report, as amended through Supplement 2 (February 1975); (3) the calculation of new source terms based on models and parameters as given in NUREG-0017 (April 1976), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)"; and (4) the calculation of the cost-benefit ratio for potential radwaste system additions, using doses based on the source terms calculated in (3) above and guidance as given in Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors" (March 1976).

The individual and population doses were calculated using the guidance in Regulatory Guide 1.109, "Calculation of Annual Average Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I" (March 1976). Atmospheric transport and diffusion estimates applicable to the dose calculations are described in Section 2.3.5 of this supplement.

We have determined the quantities of radioactive materials that are estimated to be released in the liquid and gaseous effluent streams during normal operation including anticipated operational occurrences. The principal radionuclides expected to be released in liquid and gaseous effluents are given in Tables 11.3 and 11.4 of this supplement. In making these determinations, we have considered waste flow rates,

TABLE 11.3

CALCULATED RELEASES OF RADIOACTIVE MATERIALS IN LIQUID EFFLUENTS
(Curies per year)

Corrosion/Activation Products	
Cr-51	1.4(-4)
Mn-54	1(-3)
Fe-55	1.4(-4)
Fe-59	8(-5)
Co-58	5.3(-3)
Co-60	8.9(-3)
Zr-95	1.4(-3)
Nb-95	2(-3)
Np-230	6(-5)

Fission Products	
Br-83	3(-5)
Rb-86	2(-5)
Sr-89	3(-5)
Sr-91	1(-5)
Mo-99	3.7(-2)
Tc-99m	2.3(-2)
Ru-103	1.4(-4)
Ru-106	2.4(-3)
Ag-110m	4.4(-4)
Te-127m	2(-5)
Te-127	3(-5)
Te-129m	1.1(-4)
Te-129	7(-5)
I-130	9(-5)
Te-131m	5(-5)
I-131	4.6(-2)
Te-132	1.1(-3)
I-132	2.5(-3)

Fission Products (continued)	
I-133	2.3(-2)
I-134	2(-5)
Cs-134	2.6(-2)
I-135	4.7(-3)
Cs-136	3(-3)
Cs-137	3.4(-2)
Ba-137m	9.3(-3)
Ba-140	1(-5)
Ce-144	5.2(-3)
All Others	6(-5)
Total (except H-3)	2.4(-1)
H-3	5.5(+2)

TABLE 11.4

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS
(Curies per year)

Nuclide	Waste Gas Processing System	Reactor Bldg	Auxiliary Bldg	Turbine Bldg	Condenser Air Removal Vent	Total
Kr-83m	a	a	a	a	a	a
Kr-85m	a	1	1	a	a	2
Kr-85	280	110	3	a	2	390
Kr-87	a	a	a	a	a	a
Kr-88	a	2	3	a	2	7
Kr-89	a	a	a	a	a	a
Xe-131m	12	50	2	a	a	64
Xe-133m	a	35	3	a	2	40
Xe-133	180	5600	250	a	160	6200
Xe-135m	a	a	a	a	a	a
Xe-135	a	10	5	a	3	18
Xe-137	a	a	a	a	a	a
Xe-138	a	a	a	a	a	a
I-131	a	1.3(-4)	5.5(-3)	1(-3)	3.4(-3)	1(-2)
I-133	a	1.3(-4)	5.8(-3)	1.1(-3)	3.6(-3)	1.1(-2)
Co-60	7(-5)[b]	1.2(-6)	2.7(-4)	a	a	3.4(-4)
Co-58	1.5(-4)	2.6(-6)	6.4(-4)	a	a	7.5(-4)
Fe-59	1.5(-5)	2.6(-7)	6(-5)	a	a	7.5(-5)
Mn-54	4.5(-5)	7.6(-7)	1.8(-4)	a	a	2.3(-4)
Cs-137	7.5(-5)	1.3(-6)	3(-4)	a	a	3.8(-4)
Cs-134	4.5(-5)	7.6(-7)	1.8(-4)	a	a	2.3(-4)
Sr-90	6(-7)	1(-8)	2.4(-6)	a	a	3(-6)
Sr-89	3.3(-6)	5.9(-8)	1.3(-5)	a	a	1.6(-5)
C-14	7	1	a	a	a	8
H-3	a	280	280	a	a	560
Ar-41	a	25	a	a	a	25

a - negligible compared to overall source term, e.g., less than 1.0 curie per year noble gases, less than 1(-4) curie per year iodine, less that 1 percent of total for particulates

b - exponential notation: 7(-5) = 7 x 10^{-5}

concentrations of radioactive materials, and equipment decontamination factors, which are consistent with those expected over the expected 30-year operating life of the plant, considering normal operation including anticipated operational occurrences. A list of the parameters used in these determinations is given in Table 11.5.

In our evaluation, we considered releases of radioactive materials in liquid and gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the 30-year operating life of the plant and have determined that: (1) the release of all radioactive materials above background in liquid effluents will not result in an annual dose or dose commitment to any individual in an unrestricted area from all pathways of exposure in excess of 3 millirems to the total body or 10 millirems to any organ; (2) the release of all radioactive materials above background in gaseous effluents will not result in an estimated annual air dose at any location near ground level which could be occupied by individuals in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation; and (3) the release of all radioactive iodine and radioactive material in particulate form above background will not result in an annual dose or dose commitment to any individual in an unrestricted area from all pathways of exposure in excess of 15 millirems to any organ.

We have also considered the potential effectiveness of augmenting the liquid and gaseous radwaste treatment systems using items of reasonably demonstrated technology and have determined that additional augmentation will not effect reductions in the cumulative population dose reasonably expected within a 50-mile radius of the reactor at a cost of less than $1,000 per total body man-rem or $1,000 per man-thyroid-rem. Parameters used in our evaluation are given in Table 11.8.

We evaluated potential radwaste system augments based on a study of the applicant's system designs, on the dose information provided in Tables 11.6 and 11.7 of this supplement, on the basis of an interim value of $1,000 per total body man-rem and $1,000 per man-thyroid-rem for reductions in dose by the application of augments, and on the cost of potential radwaste system augments as presented in Regulatory Guide 1.110 (March 1976). Potential radwaste system augments were selected from the list given in Regulatory Guide 1.110. We found that the Total Annualized Cost for each augment exceeded the $1,000 per total body man-rem or $1,000 per man-thyroid-rem cost-benefit ratio.

TABLE 11.5

PRINCIPAL PARAMETERS AND CONDITIONS USED IN CALCULATING RELEASES OF RADIOACTIVE MATERIAL IN LIQUID AND GASEOUS EFFLUENTS

Reactor Power Level (megawatts thermal)	2800
Plant Capacity Factor	0.80
Failed Fuel	0.12 percent (a)
Primary System	
Mass of Coolant (pounds)	7.2×10^5
Letdown Rate (gallons per minute)	45
Shim Bleed Rate (gallons per day)	1.5×10^3
Leakage to Secondary System (pounds per day)	100
Leakage to Containment Building	b
Leakage to Auxiliary Buildings (pounds per day)	160
Frequency of Degassing for Cold Shutdowns (per year)	2
Secondary System	
Steam Flow Rate (pounds per hour)	1.2×10^7
Mass of Steam/Steam Generator (pounds)	1.4×10^3
Mass of Liquid/Steam Generator (pounds)	2.7×10^4
Secondary Coolant Mass (pounds)	3×10^6
Rate of Steam Leakage to Turbine Bldg (pounds per hour)	1.7×10^3
Fraction of Feedwater Processed Through Condensate Demineralizers	0.7
Containment Building Volume (cubic feet)	2.1×10^6
Annual Frequency of Containment Purges (shutdown)	4
Annual Frequency of Containment Purges (at power)	20
Iodine Partition Factors (gas/liquid)	
Leakage to Auxiliary Building	0.0075
Steam Generator	1.0
Leakage to Turbine Building	1.0
Main Condenser/Air Ejector (volatile species)	0.15
Decontamination Factors (liquid wastes)	

	Boron Recovery System	Floor Drain Wastes, Inorganic Chemical Wastes, Regenerant Solutions	Laundry and Hot Shower Drains
I	1×10^4	1×10^4	1
Cs, Rb	2×10^4	1×10^5	1
Others	1×10^5	1×10^5	1

	All Nuclides Except Iodine	Iodine
Radwaste Evaporator DF	10^4	10^3
BRS Evaporator DF	10^3	10^2

[a]This value is constant and corresponds to 0.12 percent of the operating power fission product source term as given in NUREG-0017, April 1976.

[b]1 percent per day of the primary coolant noble gas inventory and 0.001 percent per day of the primary coolant iodine inventory.

TABLE 11.5 (Continued)

	Anions	Cs, Rb	Other Nuclides
Boron Recycle Feed Demin. DF (H_3BO_3)	10	2	10
Primary Coolant Letdown Demin. DF (Li_3BO_3)	10	2	10
Evaporator Condensate Polishing Demineralizer (H^+OH^-) DF	10	10	10
Mixed Bed Condensate Demin.	10	2	10
Turbine Air Removal System and Containment Building Internal Recirculation System Charcoal Filter DF (Iodine Removal)			10
Fuel Handling Building and Auxiliary Building Ventilation System Charcoal Filter DF (Iodine Removal)			10

TABLE 11.6

COMPARISON OF CALCULATED DOSES FROM OPERATION WITH SECTIONS II.A, II.B AND II.C OF APPENDIX I TO 10 CFR PART 50
(Doses to Maximum Individual Per Reactor Unit)

Criterion	Appendix I Dose Design Objective	Calculated Doses
Liquid Effluents		
Dose to total body from all pathways	3 millirems per year	1.7 millirems per year
Dose to any organ from all pathways	10 millirems per year	2.3 millirems per year
Noble Gas Effluents		
Gamma dose in air	10 millirads per year	0.51 millirad per year
Beta dose in air	20 millirads per year	1.5 millirads per year
Dose to total body of an individual	5 millirems per year	0.30 millirem per year
Dose to skin of an individual	15 millirems per year	1.0 millirem per year
Radioiodines and Particulates[a]		
Dose to any organ from all pathways	15 millirems per year	5.9 millirems per year

[a]Carbon-14 and tritium have been added to this category.

TABLE 11.7

CALCULATED POPULATION DOSES (MAN-REM) FOR COST-BENEFIT ANALYSIS, SECTION II.D OF APPENDIX I TO 10 CFR PART 50*

Pathway	Total Body	Thyroid
Liquid	7.4	12.0
Gaseous	3.5	4.4

*Based on the population reasonably expected to be within a 50-mile radius of the reactor.

TABLE 11.8

PRINCIPAL PARAMETERS USED IN THE COST-BENEFIT ANALYSIS

Labor Cost Correction Factor, FPC Region I	1.6
Indirect Cost Factor[a]	1.75
Cost of Money[b]	10 percent
Capital Recovery Factor[c]	0.1061

[a]From Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (March 1976)."

[b]Applicant did not provide his cost of money; we used the value of 10 percent as derived from a recent Annual Report and Prospectus.

[c]The applicant provided a value of 16 percent as his Capital Recovery Factor. The value of 16 percent is not consistent with the applicant's cost of money and a 30-year recovery period and would be more appropriate as a fixed charge rate; therefore, we have assumed a value of 0.1061 for the Capital Recovery Factor. This assumption does not change the results of our evaluation.

15.0 ACCIDENT ANALYSES

15.3 Radiological Consequences of Accidents

15.3.1 General

As noted in the Safety Evaluation Report, we had previously concluded that with a containment leak rate of 0.13 percent per day and a dose reduction factor of 6.6, the offsite dose guidelines of 10 CFR Part 100 would be met.

We have reviewed the revised spray additive system described in Section 6.2.3 of this supplement, and conclude that this system, although slightly less effective for iodine washout than the system originally proposed in that it does not remove the organic form of iodine, results in a sufficiently rapid absorption of the dominant elemental form to meet the offsite dose guidelines of 10 CFR Part 100 with a containment leak rate of 0.13 percent per day. Table 15.1 has been completed to show the potential offsite doses resulting from the postulated loss-of-coolant accident.

15.3.2 Design Basis Accident Assumptions

In the Safety Evaluation Report, we had not completed subparagraph 4 of this section covering assumptions dealing with iodine removal. Because the applicant has now provided an acceptable spray additive system, these parameters are listed below.

4. Iodine removal by the containment spray system was based on:

Sprayed containment volume	1.764×10^6 cubic feet
Unsprayed containment volume	3.950×10^5 cubic feet
Mixing rate between sprayed and unsprayed regions	2.0 turnovers of unsprayed volumes per hour plus 18,000 cubic feet per minute
Iodine removal coefficients	
Elemental	10.0 $hours^{-1}$
Organic	0
Particulate	0.4 $hours^{-1}$
Elemental Iodine decontamination factor	100

TABLE 15.1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

Accident	Two Hour Exclusion Boundary (610 Meters)		Course of Accidents Low Population Zone (3218 Meters)	
	Thyroid (rem)	Whole Body (rem)	Thyroid (rem)	Whole Body (rem)
Loss-of-Coolant	280	8.2	108	2.1
Post-LOCA Hydrogen Purge Dose			< 1	
Fuel Handling	46	3		< 1
Steam Generator Tube Rupture	6	< 1		
Steam Generator Tube Rupture with Iodine Spike	76	< 1		
Steam Line Break	2	< 1		
Loss of Offsite Power	< 1	< 1		
Loss of Offsite Power with Coincident Iodine Spike	1	< 1		
Gas Decay Tank Rupture	Negligible	6	Negligible	< 1
Rod Ejection**				
Case I	24	< 1	11	< 1
Case II	102	2	19	< 1

** Actual rod ejection doses will not exceed the doses for Case I (releases through the containment) or Case II (releases through the secondary system).

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

A Subcommittee of the Advisory Committee on Reactor Safeguards (Committee) considered the application for an operating license for Three Mile Island Unit 2 on September 23 and 24, 1976 in Harrisburg, Pennsylvania. The full Committee completed its review of the application at its 198th meeting on October 15, 1976. A copy of the Committee's report dated October 22, 1976 is attached as Appendix D. The following paragraphs discuss the current status of each item on which the Committee commented or made recommendations in that report.

(1) The Committee indicated that it believes that the proposed power distribution monitoring methods may be acceptable but that an augmented startup program should be employed, and that satisfactory experience at 100 percent steady state power and during transients at less than full power should be obtained. Further, this experience should be reviewed and evaluated by the staff prior to operating at full power in the load following mode.

Such a program was conducted at the Rancho Seco Nuclear Generating Station Unit 1 reactor, which has a core identical to that of Three Mile Island Unit 2. Rancho Seco Unit 1 was operated at 2568 megawatts thermal ($\sim$92 percent rated power) for an extended period in order to proof test the bleed and feed mode of operation prior to operation at full power. A conscious effort was made to exercise the bleed and feed system. Up-ramp and down-ramp power changes were performed in the turbine-following, reactor-following, and integrated modes of the integrated control system.

A "pseudo-design power transient" was performed in which a rapid reduction in power by 50 percent of full power was performed, power was held at the low value ($\sim$ 30 percent of full power) until the xenon buildup was maximum, and the power was then quickly raised to its original value. In addition to this pseudo-design transient, several actual load following transients were performed in response to dispatcher commands. This situation came about due to run-off conditions in the spring of 1975 that dictated the use of hydroelectric plants as base loads for the power grids.

The results of the bleed and feed operations were evaluated with respect to the accuracy with which they could be performed. Over 50 transients were performed, some with boron alone and others with combination of boron and control rod movement. The end-point boron concentration was checked against the target value and the deviation in the results was within the measurement uncertainty in the boron concentration.

Power distribution measurements were performed upon return to full power after the pseudo-design transient and showed that thermal margins were met when extrapolated to 102 percent of full power. The ability of the axial imbalance and rod position monitoring to limit power peaking was demonstrated in the tests which were used to calibrate the axial imbalance instrumentation. These tests showed margins to departure from nucleate boiling ratio and fuel melt limits when extrapolated to design overpower conditions.

Upon completion of this program, the licensee requested permission to increase power to 2772 megawatts thermal (the design rated power). This request was reviewed by the staff and the Committee and approval to proceed to 100 percent full power was granted. While the request was being reviewed, a broken turbine blade necessitated the shutdown of the reactor. In December 1975, the reactor was again started up and the rest of the startup tests (those at greater than 92.6 percent of full power) were conducted. Full power was achieved in March 1976 and operation at full power continued into April. At this time insulation failure in the station generator necessitated shutdown of the reactor.

Repairs were completed and full power was again reached in October 1976.

The ability to calculate power distributions - particularly during transients - is being pursued with Babcock & Wilcox on a generic basis. A topical report on the nuclear uncertainty factor is in the final stages of preparation at Babcock & Wilcox and reports on the methods used to establish operating limits and protection system set points are being prepared. These are expected to be received well before startup testing is initiated on Three Mile Island Unit 2.

We believe that data which will be obtained from a closely monitored normal startup testing program at Three Mile Island Unit 2, in conjunction with the information available from the augmented startup program performed on the Rancho Seco reactor with an identical core design, is sufficient to evaluate the behavior of this bleed and feed reactor at full power and during load following transients.

If the evaluation of the generic reports discussed above indicates that an augmented startup program on this plant might be desirable, we will reevaluate our position, but at this time we conclude that such an augmented startup program is not necessary.

(2) The Committee indicated that the issue of asymmetric loads on the reactor vessel and its internal structures for certain postulated loss-of-coolant accidents should be resolved in a manner satisfactory to the staff.

This item is generic in nature and as such is being pursued and monitored as part of our effort in resolving all generic items (see Appendix C, Item II C-2). The discussion in Section 3.9.3 of the Safety Evaluation Report notes that considering the design of this plant and the probability of occurrence of the event in question, we conclude that operation of this facility would be acceptable until we complete our generic review. At that time, we will consider imposition of any generic solution on Three Mile Island Unit 2 to add to the overall safety margin for the facility.

(3) The Committee noted that anticipated transients without scram remains an outstanding issue pending completion of our review of the Babcock & Wilcox generic analyses, and recommended early resolution of this matter in a manner acceptable to us.

The discussion in Section 7.2.2 of the Safety Evaluation Report notes that considering the probability of occurrence of the event in question, we conclude that limitations on operation on this account are not necessary or appropriate until such time as any facility modifications found necessary by our review of the required additional information can be implemented.

(4) The Committee stated that it believes that the details of the flood emergency plan, particularly relating to reentry into the station during the post-flood period, needs to be more clearly delineated.

We had concluded in Section 2.4.5 of the Safety Evaluation Report that the flood emergency procedures were acceptable, noting that the applicant's flood emergency plan identifies the operational procedures to be taken and lists the specific flood protection actions to be taken.

The applicant has since provided additional information to further document the adequacy of these plans. He notes that a seven-day supply of fuel for each diesel-generator is provided in a flood-protected onsite area. This supply would obviate the need to provide additional fuel during the period of about 72 hours during which the probable maximum flood may preclude normal access to the island. The access bridge is designed to survive the probable maximum flood and any necessary additional fuel could be provided when the flood level recedes. In addition, if fuel replenishment becomes necessary before normal access is possible, fuel delivery by boat or helicopter would be possible.

With regard to personnel access, the applicant states that sufficient personnel are always onsite to accomplish and maintain safe unit shutdown. If additional personnel become necessary before normal access is restored, alternate means of transportation such as boat or helicopter could be utilized.

We believe this additional information satisfies the concerns raised by the Committee.

(5) The Committee supported our program for further evaluation of plant fire protection, and recommended that high priority be given to evaluations of Three Mile Island Unit 2 and other plants nearing completion of construction to facilitate making any changes required to improve fire protection capability.

Subsequent to publication of the Safety Evaluation Report containing a discussion of fire protection in Section 9.5, we issued Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1976," which documents our position on fire protection for such plants as Three Mile Island Unit 2. On September 30, 1976, we transmitted Appendix A to the applicant and requested performance of a fire hazards analysis and a reevaluation of the fire protection program for this plant, including a comparison with Appendix A. The applicant notified us in a letter dated November 10, 1976 that the information requested in our letter would be furnished by June 1977.

We will review this information and identify any required changes as expeditiously as possible. We expect to complete the review prior to the scheduled fuel loading, and we will of course require that all necessary modifications and procedural and administrative changes be made as early as feasible to improve the capability of the fire protection system.

(6) The Committee noted that long-term post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptible to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging, and that appropriate test procedures to confirm continuous long-term seal capability should be developed.

The applicant is compiling a list of instruments for which these concerns are applicable. Although this information has not yet been submitted, we understand that those instruments with seals can be pressure tested periodically during operation. The applicant intends to present information on manufacturers' tests, proposed operational testing, and other aspects of this concern. We will review this information and report further in a supplement to this report.

(7) The Committee recommended that further review be made of the battery supplied direct current power system to assure that nonessential loads do not interfere with its safety function.

In response to this concern, the applicant has stated that all essential and nonessential loads are connected to the direct current buses through individual Class IE circuit breakers located in seismic Class I areas. Also, the fault protection coordination scheme is such that a fault within any load/feeder circuit will cause its circuit breaker to trip prior to initiation of a trip of any upstream circuit breaker. This ensures that faults in nonessential loads/feeders connected to the direct current bus do not interfere with any safety functions. In addition, the circuits in the direct current distribution panels are all individually fused on both the positive and negative sides. The fuse rating for each load in the distribution panel is such that its time-overcurrent trip characteristics for all faults will cause the fuse to open prior to the initiation of a trip of the branch circuit breaker at the direct current bus.

We have reviewed the information provided concerning this matter and conclude that the present design is consistent with other plant designs previously reviewed and found acceptable by the staff. We will require appropriate technical specifications concerning periodic testing of these circuit breaker trip characteristics to assure that the present design is adequate. We believe this information satisifies the concern raised by the Committee.

(8) The Committee recommended that further review be made to assure that no unacceptable effects such as release of hydrogen into the plant can occur from the failure of a hydrogen charging line.

The applicant has provided routing and design information on the hydrogen line showing that it passes only through a portion of the auxiliary building and not the containment. Further analyses will be submitted addressing the consequences of a break in the hydrogen line, with consideration being given to equipment available to deal with such a postulated accident. We will report further on this item in a future supplement to this report.

(9) The Committee recommended that studies be made to assure that failure of an instrument line cannot cause plant controllability problems of significance to public safety.

Analyses performed for other Babcock & Wilcox plants (Oconee) have not identified significant safety problems of plant.controllability as a result of an instrument line break. The applicant will submit an analysis examining this concern for Three Mile Island Unit 2. We will review this analysis and report further on this matter in a supplement to this report.

(10) The Committee indicated that questions regarding the management organization proposed by the applicant should be resolved to our satisfaction.

We have identified specific areas of concern to the applicant and we understand resolutions to our concerns will be proposed. When this information is submitted formally and reviewed, we will report on it in a subsequent supplement.

(11) The Committee noted that various issues related to accidents involving loss of fluid in the secondary side of the steam generator are still being reviewed.

As noted in Section 1.7 of this supplement, we have informed the applicant of our requirements for additional analyses and/or system modifications. We will report further on these matters in a subsequent supplement to the Safety Evaluation Report.

(12) The Committee recommended that additional means for evaluating the cause and likely course of various accidents be in hand prior to commercial operation to provide improved bases for timely decisions regarding offsite emergency measures.

This matter is being considered as a generic issue (Item II-11, Status Report on Generic Items. See Appendix C.) and as such will be dealt with on this plant and others when a final generic solution is developed.

(13) The Committee indicated its belief that we should further review measures to reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

We have further reviewed particular structures with regard to their possible vulnerability to sabotage, and conclude that the design, construction, and installation of these structures and the security systems at Three Mile Island Unit 2 provide an acceptable degree of security.

Further references regarding the generic aspects of this issue are provided in Appendix C. This issue is identified as Item II C-3 in the Status Report on Generic Items.

In addition, we will require submittal of an amended physical security plan in compliance with the requirements of 10 CFR Part 73.55, which is to be effective March 28, 1977.

(14) The Committee identified generic items of concern which they considered relevant to this plant, and indicated that these should be dealt with by the staff and the applicant as solutions are found.

Appendix C herein notes the disposition and status of each of the indicated items.

20.0 FINANCIAL QUALIFICATIONS

In the Safety Evaluation Report, we noted that we were reviewing the applicant's financial qualifications. The information submitted by the applicant, our completed review of that information, and the summary presented below, were based on the ownership arrangement in effect at the time, as described in Amendment 13 to the application and described briefly in Section 1.1 of the Safety Evaluation Report.

The applicant has stated that certain changes in the ownership arrangement will be made in the near future. At such time as these changes are formally defined, we will review their effect, if any, on the financial qualifications of the applicant and report further as appropriate in a later supplement to the Safety Evaluation Report.

20.1 Introduction

In accordance with the regulations noted in the Safety Evaluation Report, Metropolitan Edison Company (Metropolitan Edison), Jersey Central Power & Light Company (Jersey Central), and Pennsylvania Electric Company (Pennsylvania Electric) supplied operating and shutdown cost estimates and additional financial information for the Three Mile Island Nuclear Station, Unit 2. The following analysis summarizes our review of this financial information and addresses each owner's financial qualifications to operate and, if necessary, to permanently shut down and safely maintain the subject facility.

Metropolitan Edison Company, Jersey Central Power & Light Company, and Pennsylvania Electric Company, each of which is a wholly-owned subsidiary of General Public Utilities Corporation, supply electricity to approximately 1.5 million customers over a 24,000 square mile service area in Pennsylvania and New Jersey. Financial information for each of the applicants for the 12 months ended June 30, 1976 is presented in Table 20.1.

TABLE 20.1

FINANCIAL INFORMATION

	Metropolitan Edison	Jersey Central	Pennsylvania Electric
Revenues (millions)	$ 259.3	$ 444.4	$ 330.0
Net Income (millions)	48.0	53.8	50.7
Total Capitalization (millions)	979.4	1,367.0	1,095.9
Bond Rating			
(Moody's/S & P's	A/A	Baa/BBB+	A/BBB

Metropolitan Edison will receive a 50 percent share in the output from the Three Mile Island Unit 2 facility, with Jersey Central and Pennsylvania Electric each receiving a one-quarter interest in the plant's output.

20.2 Estimated Operating and Shutdown Costs

For the purpose of estimating the unit's annual cost, the applicant assumed January 1978 as the startup date for commercial operation of the facility. The applicant's estimate of the total annual cost of operating the unit during each of the first five years of operation is presented in Table 20.2. The unit costs (mills per kilowatt hour) are based on a net electrical capacity of 906 megawatts electric.

TABLE 20.2

ANNUAL COST OF UNIT OPERATION

	Plant Capacity	Operating Cost Estimate (thousands)	Mills per kilowatt-hour
1978	65 percent	$ 162,865	31.6
1979	56 percent	162,072	36.5
1980	64 percent	161,106	31.7
1981	67 percent	158,460	29.8
1982	71 percent	157,764	28.0

In estimating the costs of permanently shutting down the facility, the applicant considered three alternatives: dismantling, entombment, and mothballing. Dismantling the unit, which involves removing all fuel assemblies, radioactive fluids and waste, and other materials having radioactive activities above unrestricted activity levels, is estimated to cost approximately $117 million. Expenditures for entombment are projected to be $45 million initially, with an annual surveillance expense of $68,000 thereafter. Entombment consists of sealing all remaining highly radioactive components within a biologically secure structure after having removed all fuel assemblies and radioactive fluids and waste. The estimated expense of putting the facility in a state of protective storage, or mothballing the unit, is $6.8 million, with an additional expenditure of $200,000 annually for radiation monitoring, environmental surveillance, and appropriate security procedures. All decommissioning costs are stated in 1975 dollars.

20.3 Source of Funds

Metropolitan Edison, Jersey Central, and Pennsylvania Electric expect to cover all operating expenses, including taxes, and interest payments through revenues generated from their system-wide sales of electricity. The owners have consistently exhibited the ability to cover all operating expenses as evidenced by the revenue to expense ratio presented in Table 20.3. The staff assumes that shutdown and subsequent maintenance costs will either be expensed in the year incurred or amortized over a period of years, depending on the rate-making policy of the regulatory authorities.

TABLE 20.3

RATIO OF OPERATING REVENUE TO OPERATING AND INTEREST EXPENSES

	Metropolitan Edison	Jersey Central	Pennsylvania Electric
1975	1.24	1.15	1.19
1974	1.26	1.21	1.19
1973	1.26	1.23	1.24
1972	1.26	1.27	1.20
1971	1.22	1.21	1.23
1966-1970 (average)	1.19	1.23	1.27

During 1975, Metropolitan Edison, Jersey Central, and Pennsylvania Electric sold electricity for an average unit price (mills per kilowatt hour) of 37.3, 39.9, and 32.4, respectively. Even without applying a growth factor to these 1975 prices, they are still significantly above the 1978 estimated unit cost of generating electricity at Three Mile Island Unit 2 (Table 20.2).

20.4 Conclusion

In accordance with the regulations cited in the Safety Evaluation Report, there must be reasonable assurance that the owners can obtain the necessary funds to cover the estimated costs of the activities contemplated under the license. Based on our analysis, we have concluded that Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company satisfy this reasonable assurance standard and, therefore are financially qualified to operate and, if necessary, shut down and safely maintain Three Mile Island Nuclear Station Unit 2. Our conclusion is supported by the following factors as discussed above: (1) the ability to earn revenues sufficient to cover all operating expenses, including interest payments and taxes; and (2) the projected output of lower unit cost electricity from this facility, as compared with the utilities' present average price of electricity.

APPENDIX A

CHRONOLOGY OF OPERATING LICENSE STAGE RADIOLOGICAL SAFETY REVIEW

The following updating of the chronology is provided.

August 6, 1976	Letter from applicant on dike repair.
August 31, 1976	Letter from applicant on reactor vessel support analysis.
September 7, 1976	Letter from applicant transmitting Amendment 44.
September 8, 1976	Letter from applicant transmitting Amendment 45.
September 13, 1976	Meeting with applicant to discuss open items.
September 17, 1976	Safety Evaluation Report issued.
September 23 and 24, 1976	Meeting of subcommittee of Advisory Committee on Reactor Safeguards.
September 30, 1976	Letter from applicant transmitting Amendment 46.
September 30, 1976	Letter from applicant transmitting Amendment 47.
October 6, 1976	Meeting with applicant on open itmes.
October 15, 1976	Meeting of Advisory Committee on Reactor Safeguards.
October 22, 1976	Report of Advisory Committee on Reactor Safeguards.
November 9, 1976	Meeting with applicant on open items.
November 10, 1976	Letter from applicant on information on fire protection.
November 15, 1976	Letter from applicant transmitting Amendment 48.
November 30, 1976	Letter from applicant transmitting Amendment 49.
December 8, 1976	Letter from applicant transmitting Amendment 50.

December 20, 1976	Letter to applicant on fire protection.
December 20, 1976	Letter to applicant transmitting letter to Babcock & Wilcox on Appendix K evaluation.
January 5, 1976	Letter to applicant transmitting request for additional information.
January 21, 1977	Letter from applicant furnishing information on Appendix K evaluation.
January 24, 1977	Meeting with applicant on change of ownership percentages.
January 26, 1977	Meeting with applicant on operating organization.

APPENDIX B

BIBLIOGRAPHY FOR SAFETY EVALUATION REPORT

The following additional references are provided.

Meteorology

77. Sagendorf, J. F., and J. T. Goll, 1976: XOQDOQ, Program for the Evaluation of Routine Effluent Releases at Nuclear Power Stations, (DRAFT). U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D. C.

Core Performance

78. Rancho Seco Unit 1 Startup Report, March 1975 (Docket 50-312).

79. Rancho Seco Unit 1 Performance Report, March 1975 (Docket 50-312).

80. Letter, Sacramento Municipal Utility District (Mattimoe) to the Nuclear Regulatory Commission (Engleken), dated July 30, 1976.

81. Letter, W. Kerr to W. A. Anders, dated July 16, 1975, on proceedings of the 183rd meeting of the ACRS, July 10 - 12, 1975.

Generic Items

82. Letter, B. C. Rusche to M. Bender, dated January 31, 1977 transmitting the staff Status Report on Generic Items.

APPENDIX C

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS-GENERIC MATTERS

The Advisory Committee on Reactor Safeguards (Committee) periodically issues a report listing various generic items applicable to large lightwater reactors. These are items which we and the Committee, while finding present plant designs acceptable, believe have the potential of adding to the overall safety margin of nuclear power plants, and as such should be considered for application to the extent reasonable and practicable as solutions are found, recognizing that such solutions may occur after completion of the plant. This is consistent with our continuing efforts toward reducing still further the already small risk to the public health and safety from nuclear power plants. The Committee report concerning these generic items on which this Appendix is based was issued to the Commission on April 16, 1976 in a letter from Committee Chairman D. Moeller to Commission Chairman M. Rowden.

The status of staff efforts leading to resolution of all these generic matters is contained in our Status Report on Generic Items periodically transmitted to the Committee. The latest such Status Report is contained in a letter from B. Rusche to M. Bender dated January 31, 1977.

The Committee in its report on Three Mile Island Unit 2 dated October 15, 1976, identified which of these generic items it deems applicable to Three Mile Island Unit 2. For many of the items so identified, we have provided in the Safety Evaluation Report specific discussions particularizing for Three Mile Island Unit 2 the generic status given in the Status Report.

These items are listed below with the appropriate section numbers of the Safety Evaluation Report and/or this supplement where such discussions are to be found. The numbering corresponds to that in the April 16, 1976 report of the Committee.

For those items applicable to Three Mile Island Unit 2 which have not yet progressed to where specific action can be initiated relevant to individual plants, our Status Report on Generic Items referred to above provides the appropriate information.

Group II

1. Turbine Missiles - Section 3.5.1

2. Effective Operation of Containment Sprays in a LOCA - Sections 6.2.3 and 15.3.1

4. Instruments to Detect Fuel Failures - Section 9.3.3

5. Monitoring for Excessive Vibration or Loose Parts Inside the Pressure Vessel - Section 5.6

6. Common Mode Failures - Section 7.2.2

Group IIA

1. Pressure in Containment Following a LOCA - Section 6.2.1

4. Rupture of High Pressure Lines Outside Containment - Section 3.6

5. PWR Pump Overspeed During a LOCA - Section 5.5.1

6. Isolation of Low Pressure from High Pressure Systems - Section 7.4.3

7. Steam Generator Tube Failures - Section 5.5.2

Group IIC

1. Locking Out of ECCS Power-Operated Valves - Section 7.6.2

2. Fire Protection - Section 9.5 and Section 18, Item 5

3. Design Features to Control Sabotage - Section 18, Item 13

5. Vessel Support Structures - Section 3.9.3

6. Water Hammer - Section 6.3.2. In addition, the principal area of concern in this item has been the feed inlet to the steam generators. This has not been a problem in operating Babcock & Wilcox plants because of system and component design and is not expected to be a concern in Three Mile Island Unit 2.

APPENDIX D

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D. C. 20555

October 22, 1976

Honorable Marcus A. Rowden
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON THREE MILE ISLAND NUCLEAR STATION, UNIT 2

Dear Mr. Rowden:

During its 198th meeting, October 14-16, 1976, the Advisory Committee on Reactor Safeguards completed its review of the application of the Metropolitan Edison Company, Jersey Central Power and Light Company, and Pennsylvania Electric Company (Applicants) for a license to operate Three Mile Island Nuclear Station, Unit 2. This project was also considered during a Subcommittee meeting held in Harrisburg, Pennsylvania, on September 23 and 24, 1976. Members of the Committee visited the facility on September 23, 1976. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, General Public Utilities Service Corporation, the Babcock and Wilcox Company (B&W), Burns and Rowe, Inc., and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had available the documents listed below. The Committee reported on the application for a construction permit for Unit 1 on January 17 and April 12, 1968, and for an operating license for Unit 1 on August 14, 1973. The Committee reported on the application for a construction permit for Unit 2 on July 17, 1969.

The Three Mile Island Nuclear Station, Units 1 and 2, is located on Three Mile Island near the eastern shore of the Susquehanna River, about 12 miles southeast of Harrisburg, Pennsylvania. About 2380 people live within a two-mile radius of the site (the low population zone). The minimum exclusion distance is 2000 feet. The nearest population center is Harrisburg (1970 population 68,000).

Several changes have been made to bring the Babcock and Wilcox Emergency Core Cooling System (ECCS) evaluation model into conformance with the requirements of 10 CFR 50.46, and Appendix K to Part 50. Analyses of a spectrum of break sizes appropriate to Three Mile Island, Unit 2 have been completed using the approved B&W generic evaluation model. The results of the analyses for the reactor coolant pump discharge break, believed to be the "worst" break, show maximum allowable linear heat generation rates as a function of elevation in the reactor core ranging from 15.5 to 18.0 kilowatts per foot. Corresponding calculated post-accident peak clad temperatures range from 2002°F to 2146°F. The NRC Staff has identified additional information that it will require to complete its review and the Applicants' submittal is expected by the end of 1976. The Applicants propose to use both in-core and ex-core instrumentation to assure accuracy of measurement of core power distributions. The Committee believes that the proposed monitoring methods may be acceptable, but that an augmented startup program should be employed, and that satisfactory experience at 100% steady state power and during transients at less than full power should be obtained. This experience should be reviewed and evaluated by the NRC Staff prior to operating at up to full power in a load following mode. The Committee wishes to be kept informed.

A question has arisen concerning asymmetric loads on the reactor vessel and its internal structures for certain postulated loss-of-coolant accidents in pressurized water reactors. The Staff has required the Applicants to supply further information in order to complete its assessment of this matter. This issue should be resolved in a manner satisfactory to the NRC Staff.

The question of whether Unit 2 requires design modifications in order to comply with WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors", remains an outstanding issue pending the NRC Staff's completion of its review of B&W generic analyses of anticipated transients without scram. The Committee recommends that the NRC Staff, the Applicants and B&W continue to strive for an early resolution of this matter in a manner acceptable to the NRC Staff. The Committee wishes to be kept informed.

Emergency plans have been developed to allow plant shutdown and maintenance of safe shutdown in the event of a maximum probable flood. Such a postulated flood would top the levee surrounding the plant by several feet. Included in the plan is the fastening of water tight steel panels in doorways and other openings of safety related structures. The Committee believes that the details of this plan, particularly relating to re-entry into the station during the post-flood period, need to be more clearly delineated.

The Committee supports the NRC Staff's program for evaluation of fire protection in accordance with Branch Technical Position APCSB 9.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants". The Committee recommends that the NRC Staff give high priority to the completion of both owner and Staff evaluations and to recommendations for Three Mile Island Unit 2 and other plants nearing completion of construction in order to maximize the opportunity for improving fire protection while areas are still accessible and changes are more feasible.

The Committee notes that long-term post-accident operation of the plant to maintain safe shutdown conditions may be dependent on instrumentation and electrical equipment within containment which is susceptable to ingress of steam or water if the hermetic seals are either initially defective or should become defective as a result of damage or aging. The Committee believes that appropriate test procedures to confirm continuous long-term seal capability should be developed.

The Committee recommends that further review be made of the battery supplied DC power system to assure that non-essential loads do not interfere with its safety function. The Committee recommends that further review be made to assure no unacceptable effects such as release of hydrogen into the plant can occur from the failure of a hydrogen charging line. The Committee also recommends that studies be made to assure that failure of an instrument line cannot cause plant controllability problems of significance to public safety.

The management organization proposed by the Applicants to delineate the safety related responsibilities of the off-site and on-site personnel of the Three Mile Island Station left open questions as to how these responsibilities are to be discharged during normal working hours and during evening, night, and weekend shifts. This matter should be resolved to the satisfaction of the NRC Staff.

The NRC Staff is still reviewing various issues related to accidents leading to loss of fluid in the steam generator secondary side, such as steam line breaks. The Committee wishes to be kept informed of the resolution of these issues.

The Committee recommends that, prior to commercial power operation of Three Mile Island Unit 2, additional means for evaluating the cause and likely course of various accidents, including those of very low probability, should be in hand in order to provide improved bases for timely decisions concerning possible off-site emergency measures. The Committee wishes to be kept informed.

The Committee believes that the Applicants and the NRC Staff should further review the Three Mile Island Nuclear Station for measures that could significantly reduce the possibility and consequences of sabotage, and that such measures should be implemented where practical.

Other generic problems relating to large water reactors are discussed in the Committee's report entitled "Status of Generic Items Relating to Light Water Reactors: Report No. 4", dated April 16, 1976. Those problems relevant to the Three Mile Island Station should be dealt with appropriately by the NRC Staff and the Applicants as solutions are found. The relevant items are: II - 1, 2, 3, 4, 5, 6, 7, 9, 11; IIA - 1, 4, 5, 6, 7, 8; IIC - 1, 2, 3, 4, 5, 6, 7.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and pre-operational testing, there is reasonable assurance that Three Mile Island Nuclear Station, Unit 2 can be operated at power levels up to 2772 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

References

1. Three Mile Island Nuclear Station, Unit 2 Final Safety Analysis Report (April, 1974) with Amendments 1 through 44.

2. Safety Evaluation Report (NUREG-0107) related to operation of Three Mile Island Nuclear Station, Unit 2, dated September, 1976.