DCR 016

May 14, 1987

Docket No. 50-289	DISTRIBUTION	Docket File
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SUBJECT: AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-50 (TAC 55750)

The Commission has issued the enclosed Amendment No. 129 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). This amendment consists of changes to the Technical Specifications (TSs) in response to your letter dated May 12, 1986, as supplemented on September 11, 1986.

In accordance with Generic Letter 83-43, the Amendment revises TMI-1 TS reporting requirements to conform with the 10 CFR 50.72 and 50.73 rule changes. Additionally, administrative and editorial changes have been made to correct, improve and clarify existing TS, this included incorporating elements of the Standard TS where appropriate.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

Gordon E. Edison, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 129 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

MY wanted revision to SE / check STATE & SECY before issuance

PDI-4	PDI-4	~~PDI-4~~	OGC
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5/14/87	5/14/87	5/14/87	5/7/87

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WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER AND LIGHT COMPANY

PENNSYLVANIA ELECTRIC COMPANY

GPU NUCLEAR CORPORATION

DOCKET NO. 50-289

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
License No. DPR-50

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by GPU Nuclear Corporation, et al. (the licensees) dated May 12, 1986, as supplemented September 11, 1986, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-50 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 129, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects I/II

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 129

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
i	i
iii	iii
iv	iv
v	v
--	1-7a
3-12	3-12
3-46	3-46
3-88	3-88
3-95	3-95
3-107	3-107
3-108	3-108
3-109	3-109
3-112	3-112
--	3-112a
3-113	3-113
3-114	3-114
3-118	3-118
3-119	3-119
--	3-119a
4-35	4-35
4-35a	4-35a
4-36	4-36
4-37	4-37
4-38	--
--	4-56
4-57	--
4-81	4-81
4-83	4-83
4-85	4-85
6-4	6-4
6-10	6-10
6-11	6-11
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6-21	6-21

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1.21 REPORTABLE EVENTS

A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

3.1.6 LEAKAGE

Applicability

Applies to reactor coolant leakage from the reactor coolant system and the makeup and purification system

Objective

To assure that any reactor coolant leakage does not compromise the safe operation of the facility.

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (excluding normal evaporative losses) exceeds one gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be placed in hot shutdown within 24 hours of detection.

3.1.6.3 If primary-to-secondary leakage through the steam generator tubes exceeds 1 gpm total for both steam generators, the reactor shall be placed in cold shutdown within 36 hours of detection.

3.1.6.4 If any reactor coolant leakage exists through a nonisolable fault in an RCS strength boundary (such as the reactor vessel, piping, valve body, etc., except the steam generator tubes), the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.5 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, 3.1.6.3, or 3.1.6.4, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case.

3.1.6.6 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within four hours of detection. The nature, as well as the magnitude, of the leak shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10 CFR 20.

3.1.6.7 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, 3.1.6.3 or 3.1.6.4, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.8 When the reactor is critical and above 2 percent power, two reactor coolant leak detection systems of different operating principles shall be in operation for the Reactor Building with one of the two systems sensitive to radioactivity. The systems sensitive to radioactivity may be out-of-service for no more than 72 hours provided a sample is taken of the Reactor Building atmosphere every eight hours and analyzed for radioactivity and two other means are available to detect leakage.

3.9 DELETED

3.10 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.10.1 The source leakage test performed pursuant to Specification 4.13 shall be capable of detecting the presence of 0.005 μCi of radioactive material on the test sample. If the test reveals the presence of 0.005 μCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations; and a Special Report of the test results that show the presence of $\geq$ .005 μCi of removable contamination shall be prepared and submitted to the NRC Region I Administrator within 90 days after completion of the test. Sealed sources are exempt from such leak tests when the source contains 100 μCi or less of beta and/or gamma emitting material or 5 μCi or less of alpha emitting material.

3.10.2 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

Bases

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, are based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

3.18.2 FIRE SUPPRESSION WATER SYSTEM

Applicability: All operating conditions

Objective: To insure adequate fire suppression capability

Specification:

3.18.2.1 The Fire Suppression Water System shall be operable with:

a. Two (2) high pressure pumps of the following four (4), shall be operable with their discharge aligned to the fire suppression header and automatic initiation logic operable. Any two of the pumps provide combined capacity greater than 3575 gal/min:

1. Circulating Water Flume Diesel Fire Pump
2. River Water Diesel Fire Pump, Unit 1
3. River Water Diesel Fire Pump, Unit 2
4. River Water Motor Fire Pump, Unit 1

b. Two (2) separate water supplies of the following four (4) each containing a minimum of 90,000 gallons:

1. Altitude Tank
2. Circulating Water Flume
3. Unit 1 River Water Intake
4. Unit II River Water Intake

c. An operable flow path capable of taking suction from two of the operable sources listed in b, above, and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant curb valves and the front valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

Action:

3.18.2.2 a. With less than the above required equipment OPERABLE, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the NRC Region I Administrator within the next 30 days outlining the plans and procedures to be used to provide for loss of redundancy in this system.

b. With the FIRE SUPPRESSION WATER SYSTEM INOPERABLE:

1. Establish a backup FIRE SUPPRESSION WATER SYSTEM within 24 hours, or
2. Be in hot shutdown within 1 hour and cold shutdown within the next 30 hours.

Amendment No. 32, 129
(11/30/77)

3.19 CONTAINMENT SYSTEMS

3.19.1 CONTAINMENT STRUCTURAL INTEGRITY

Applicability: Applies to the structural integrity of the reactor building.

OBJECTIVE: To define the inservice tendon surveillance program for the reactor building prestressing system.

Specification

3.19.1.1 With the structural integrity of the containment not conforming to the requirements of 4.4.2.1.1.b, perform an engineering evaluation of the structural integrity of the containment to determine if COLD SHUTDOWN is required. The margins available in the containment design may be considered during the investigation. If the acceptability of the containment tendons cannot be established within 48 hours, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.19.1.2 With the structural integrity of the containment otherwise not conforming to the requirements of Specification 4.4.2.1, the condition is potentially reportable in accordance with 10 CFR Part 50.72 and 50.73 per 4.4.2.1.6.b.

RADIOACTIVE EFFLUENTS

LIQUID EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.22.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released from the unit to the site boundary (see Figure 5-4) shall be limited:

a. During any calendar quarter to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and

b. During any calendar year to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.

APPLICABILITY: At all times

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the NRC Region I Administrator within 30 days, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent 3 calendar quarters so that the cumulative dose or dose commitment to any individual from such releases during these four calendar quarters is within 3 mrem to the total body and 10 mrem to any organ. This Special Report shall also include (1) the result of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.

BASES

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable". Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations in the ODCM implement the requirements in Section III.A. of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977, and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April, 1977. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.113.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.22.1.3 The appropriate portions of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the unit to unrestricted areas (see Figure 5-4) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ in any calendar month.

APPLICABILITY: At all times

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits, prepare and submit to the NRC Region I Administrator within 30 days, a Special Report which includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. A summary description of action(s) taken to prevent a recurrence.

BASES

The requirement that the appropriate portions of this system be used, when specified, provides assurance that the releases of radioactive materials in liquid effluents will be kept as low as is reasonably achievable. This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The intent of Section II.D. is to reduce effluents to as low as is reasonably achievable in a cost effective manner. This LCO satisfies this intent by establishing a dose limit which is a small fraction (25%) of Section II.A of Appendix I, 10 CFR Part 50 dose requirements. This margin, a factor of 4, constitutes a reasonable reduction.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

DOSE-NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.22.2.2 The air dose due to noble gases released in gaseous effluents from the unit to areas at and beyond the site boundary (see Figure 5-3) shall be limited to the following:

a. During any calendar quarter: $\leq$ 5 mrad for gamma radiation and $\leq$ 10 mrad for beta radiation, and

b. During any calendar year: $\leq$ 10 mrad for gamma radiation and $\leq$ 20 mrad for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the NRC Region I Administrator within 30 days, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

BASES

This specification applies to the release of radioactive materials in gaseous effluents from TMI-1.

This specification is provided to implement the requirements of Section II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through the appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Release of Reactor

Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions. NUREG-0133 provides methods for dose calculations consistent with Regulatory Guides 1.109 and 1.111.

RADIOACTIVE EFFLUENTS

GASEOUS EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.22.2.3 The dose to an individual from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents released from the unit to areas at and beyond the site boundary (See Figure 5-3) shall be limited to the following:

a. During any calendar quarter: $\leq$ 7.5 mrem to any organ, and

b. During any calendar year: $\leq$ 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the NRC Region I Administrator within 30 days, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

BASES

This specification applies to the release of radioactive materials in gaseous effluents from TMI-1.

This specification is provided to implement the requirements of Section II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statement provides the required operating flexibility and at the same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October, 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July, 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3.22.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.22.3.1 The solid radwaste system shall be used in accordance with the Process Control Program (PCP) to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times

ACTION:

a. With the provisions of the PCP not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

BASES

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50.

RADIOACTIVE EFFLUENTS

3.22.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.22.4 The annual (calendar year) dose or dose commitment to any member of the public, due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ except the thyroid, which shall be limited to less than or equal to 75 mrem.

APPLICABILITY: At all times.

ACTION:

With the calculated dose from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.22.1.2.a, 3.22.1.2.b, 3.22.2.2.a, 3.22.2.2.b, 3.22.2.3.a, or 3.22.2.3.b, calculations should be made including direct radiation contributions from the unit and from outside storage tanks to determine whether the above limits of Specification 3.22.4 have been exceeded. If such is the case, prepare and submit to the NRC Region I Administrator within 30 days, a Special Report which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceed the above limits, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

BASES

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. This specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.22.1.1 and 3.22.2.1. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

4.4.2 Structural Integrity

Specification

4.4.2.1 Inservice Tendon Surveillance Requirements

The surveillance program for structural integrity and corrosion protection conforms to the recommendations of the U.S. NRC Regulatory Guide 1.35, proposed Revision 3, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures." The detailed surveillance program for the prestressing system tendons shall be based on periodic inspection and mechanical tests to be performed on selected tendons, as specified hereafter.

4.4.2.1.1 Containment Tendons

Tendon surveillance was completed for one, three and five years following initial structural integrity using a Tech. Spec. based on Regulatory Guide 1.35 Rev. 1. The containment tendon structural integrity shall be demonstrated at five year intervals thereafter by:

a. Determining that for a representative sample* of at least 23 tendons (6 dome, 7 vertical, and 10 hoop) each tendon has a lift off force equalling, or exceeding, its lower limit predicted for the time of the test as defined in NRC Regulatory Guide 1.35, "Inservice Inspection for Ungrouted Tendons in Prestressed Concrete Containments", Proposed Revision 3, April, 1979.

   If the lift off force of a selected tendon in a group lies between the prescribed lower limit and 90% of that limit, one tendon on each side of this tendon shall be checked for their lift off forces. If the lift off forces of the adjacent tendons are equal to, or greater than, their prescribed lower limits at the time of the test, the single deficiency shall be considered unique and acceptable.

   If the lift off force of any one tendon lies below 90% of its prescribed lower limit, the tendon shall be considered a defective tendon. It shall be completely detensioned and a determination made as to the cause of the occurrence.

   If the inspections performed at one, three, and five years indicate no abnormal degradation of the post-tensioning system, the number of tendons checked for lift off force during subsequent tests may be reduced to a representative sample of at least 11 tendons (3 dome, 3 vertical, and 5 hoop).

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.

b. Determining that the average of the normalized* tendon lift off forces for each tendon group (vertical, dome, and hoop) is equal to, of greater than 1010 Kips for vertical tendons, 1040 Kips for dome tendons, and 1121 Kips for hoop tendons. If this requirement is not met, an additional sample of 4%, with a minimum of four and a maximum of ten, of the same group of tendons shall be inspected. If the total population of each group of the sampled tendons meets the criteria above, the structural integrity of the containment shall be considered acceptable.

c. Detensioning one tendon in each group (dome, vertical and hoop) from the representative sample. One wire shall be removed from each detensioned tendon and examined to determine:

   1. That over the entire length of the wire, the tendon wires have not undergone corrosion, cracks, or damage beyond that which was originally recorded and the extent of corrosion is within specified acceptable limits.

   2. A minimum tensile strength value of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire samples (one from each end and one at mid-length) cut from each removed wire.

   Upon retensioning, the elongation shall be within plus or minus 5% of that recorded at original stressing of the tendon. If the 5% limit is not met, an investigation shall be made to determine if wire failure is the cause.

d. Determining for each tendon in the above representative sample, that the sheathing filler grease is within acceptable limits as to:

   1. presence of voids.
   2. presence of free water.
   3. chemical and physical properties.

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* In order for the tendon lift off forces to be indicative of the average level of prestress, each lift off force is adjusted for differences which exist among the tendons due to initial lock off force and elastic shortening loss.

4.4.2.1.2 End Anchorages and Adjacent Concrete Surfaces

The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.4.2.1.1 and the adjacent concrete surfaces shall be determined through visual inspection. The condition of the end anchorage and adjacent concrete shall be recorded. The acceptance criteria shall be that all crack widths greater than 0.010 inch shall be recorded and evaluated. Any crack width greater than 0.050 inch shall be cause for investigation to determine the amount of structural impairment upon the reactor building and its continued structural integrity. Changes in the condition of the end anchorage or the concrete from that previously recorded shall be noted on the record.

4.4.2.1.3 Containment Surfaces

The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (specification 4.4.1.1) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test. Any abnormal degradation must be documented.

4.4.2.1.4 Tendon Surveillance Previous Inspections

The tendon surveillance shall include the reexamination of all abnormalities (i.e., concrete scaling, cracking, grease leakage, etc.) discovered in the previous inspection to determine whether conditions have stabilized. The inspection program shall be modified accordingly if obvious deteriorating conditions are observed.

4.4.2.1.5 Inspection for Crack Growth at Dome Tendons in the Ring Girder Anchorage Areas

Concrete around the dome tendon anchorage areas shall be inspected for crack growth during ten and 15 year inspections by monitoring cracks greater than 0.005 inch in width. Select as a minimum nine dome tendon anchoring areas having concrete cracks with crack widths 0.005 inch. In the selection of dome tendon anchoring areas to be monitored, preference shall be given to those areas having cracks greater than 0.005 inch in width. The width, depth (if depths can be measured with simple existing plant instruments, (i.e., feeler gauges, wires) and length of the selected cracks shall be measured and mapped by charting. This inspection may be discontinued, if the concrete cracks show no sign of growth. If, however, these inspections indicate crack growth, an investigation of the causes and safety impact should be performed.

4.4.2.1.6 Reports

a. Within 3 months after the completion of each tendon surveillance a special report shall be submitted to the NRC Region I Administrator. This Report will include a section dealing with trends for the rate of prestress loss as compared to the predicted rate for the duration of the plant life (after an adequate number of surveillances have been completed).

b. Reports submitted in accordance with 10 CFR 50.73 shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and any corrective actions taken.

4.4.3 DELETED

4.13 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources previously subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed source shall be tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4.14 DELETED

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b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by the kinetic expansion process, of all tubes exceeding the repair limit and all tubes containing throughwall cracks) required by Table 4.19.2.

4.19.5 Reports

a. Following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired or removed from service in each steam generator shall be reported to the NRC within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be reported to the NRC within 12 months following completion of the inspection. This report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes repaired or removed from service.

c. Results of steam generator tube inspections which fall into Category C-3 require notification in accordance with 10 CFR 50.72 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence in accordance with 10 CFR 50.73.

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3 on the first sample inspection (See Table 4.19.2), these results will be reported to NRC pursuant to the requirements of Specification 4.19.5.C. Such cases will be considered by the NRC on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.

TABLE 4.19-2
STEAM GENERATION TUBE INSPECTION(2)

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.(1)	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug or repair defective tubes.
			C-3	Perform action for C-3 result of first sample.	C-3	Perform action for C-3 result of first sample.
	C-3	Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in other S.G. Provide notification to NRC pursuant to 10CFR50.72.b.2.i and submit a report pursuant to 10CFR50.73.-a.2.ii.	Other S.G. is C-1	None	N/A	N/A
			Other S.G. is C-2	Perform action for C-2 result of second sample	N/A	N/A
			Other S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Provide notification to NRC pursuant to 10CFR50.72 b.2.i and submit a report pursuant to 10CFR50.73.a.2.ii.	N/A	N/A

Notes: (1) $S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(2) For tubes inspected pursuant to 4.19.2.a.4: No action is required for C-1 results. For C-2 results in one or both steam generators plug or repair defective tubes. For C-3 results in one or both steam generators, plug or repair defective tubes and provide notification to NRC pursuant to 10 CFR 50.72.b.2.i followed by a written report pursuant to 10 CFR 50.73.a.2.ii.

The Vice President of each division within GPU Nuclear Corporation as indicated in Figure 6-1, shall be responsible for ensuring the preparation, review, and approval of documents required by the activities described in 6.5.1.1 through 6.5.1.5 within his functional area of responsibility as assigned in the GPUN Review and Approval Matrix. Implementing approvals shall be performed at the cognizant manager level or above.

ACTIVITIES

6.5.1.1 Each procedure required by Technical Specification 6.8 and other procedures including those for tests and experiments which are important to safety, and changes thereto which are important to safety, shall be prepared by a designated individual(s)/group knowledgeable in the area affected by the procedure. Each such procedure, and change thereto, shall be reviewed for adequacy by an individual(s)/group other than the preparer, but who may be from the same organization as the individual who prepared the procedure or change.

6.5.1.2 Proposed changes to the Appendix "A" Technical Specifications shall be reviewed by a knowledgeable individual(s)/group other than the individual(s) group who prepared the change.

6.5.1.3 Proposed modifications to unit structures, systems and components important to safety shall be designed by an individual/organization knowledgeable in the areas affected by the proposed modification. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification but may be from the same division as the individual who designed the modification.

6.5.1.4 Proposed tests and experiments that are important to safety shall be reviewed by a knowledgeable individual(s)/group other than the preparer but who may be from the same division as the individual who prepared the tests and experiments.

6.5.1.5 Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence, shall be reviewed by a knowledgeable individual(s)/group other than the individual/group which performed the investigation.

6.5.1.6 All REPORTABLE EVENTS shall be reviewed by an individual/group other than the individual/group which prepared the report.

6.5.1.7 Special reviews, investigations or analyses and reports thereon as requested by the Vice President TMI-1 shall be performed by a knowledgeable individual(s)/group.

6.5.1.8 The Security Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR 50, and

b. Each REPORTABLE EVENT shall undergo an independent safety review pursuant to Specification 6.5.2.5.d.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a safety limit is violated:

a. The reactor shall be shutdown and operation shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. An immediate report shall be made to the Operations and Maintenance Director, and Vice President TMI-1, and the event shall be reported to NRC in accordance with 10 CFR 50.72.

c. A complete analysis of the circumstances leading up to and resulting from the occurrence shall be prepared by the unit staff. This report shall include analysis of the effects of the occurrence and recommendations concerning operation of the unit and prevention of recurrence. This report shall be submitted to the Operations and Maintenance Director and the Vice President, TMI-1. The safety limit violation report shall be submitted to NRC in accordance with 10 CFR 50.73.

6.8 PROCEDURES

6.8.1 Written procedures important to safety shall be established, implemented and maintained covering the items referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Surveillance and test activities of equipment important to safety and radioactive waste management of equipment.

c. Refueling Operations.

d. Security Plan Implementation.

e. Fire Protection Program Implementation.

f. Emergency Plan Implementation.

g. Process Control Program Implementation.

h. Offsite Dose Calculation Manual Implementation.

i. Quality Assurance Program for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15.

j. Plant Staff Overtime, to limit the amount worked by staff performing safety-related functions in accordance with NRC Policy Statement on working hours (Generic Letter No. 82-12).

6.8.2 Further, each procedure required by 6.8.1 above, and changes thereto which are important to safety, shall be reviewed and approved as described in 6.5.1 prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered;

b. The change is approved by two members of GPUNC Management Staff qualified in accordance with 6.5.1.14 and knowledgeable in the area affected by the procedure. For changes which may affect the operational status of unit systems or equipment, at least one of these individuals shall be a member of unit management or supervision holding a Senior Reactor Operator's License on the unit.

c. The change is documented, reviewed and approved as described in 6.5.1 within 14 days of implementation.

2. The following information on aircraft movements at the Harrisburg International Airport:

   a. The total number of aircraft movements (takeoffs and landings) at the Harrisburg International Airport for the previous twelve-month period.

   b. The total number of movements of aircraft larger than 200,000 pounds at the Harrisburg International Airport for the previous twelve-month period, broken down into scheduled and non-scheduled (including military) takeoffs and landings, based on a current estimate provided by the airport manager or his designee.

3. The following information from the periodic Leak Reduction Program tests shall be reported:

   a. Results of leakage measurements,
   b. Results of visual inspections, and
   c. Maintenance undertaken as a result of Leakage Reduction Program tests or inspections.

4. The following information regarding pressurizer power operated relief valve and pressurizer safety valve challenges shall be reported:

   a. Date and time of incident,
   b. Description of occurrence, and
   c. Corrective measures taken if incident resulted from an equipment failure.

5. The following information regarding the results of specfic activity analysis in which the primary coolant exceeded limits of Technical Specification 3.1.4.1 shall be reported:

   a. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

   b. Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations;

   c. Cleanup system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

   d. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and

   e. The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

C. <u>Monthly Operating Reports.</u> Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, no later than the fifteenth of each month following the calendar month covered by the report.

6.9.2 DELETED

6.9.3 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.3.1 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

6.9.3.2 The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of the land use censuses required by Technical Specification 3.23.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and environmental radiation measurements required by Table 3.23-1 taken during the period pursuant to the locations specified in the Table and Figures in the ODCM as well as summarized and tabulated results of these analyses and measurements in a format similar to the Radiological Assessment Branch Technical Position, Revision 1, November, 1979. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map(s) of all sampling locations keyed to a table giving distances and directions from a point that is midway between the Reactor Buildings of TMI-1 and TMI-2; and the results of licensee participation in the Interlaboratory Comparison Program, required by Technical Specification 3.23.3; discussion of all deviations from the sampling schedule of Table 3.23-1; discussion of all required analyses in which the LLD required by Table 4.23-1 was not achievable.

*A single submittal may be made for the station.

6.9.4 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

Note: A single submittal may be made for the station. The submittal should combine those sections that are common to both units at the station however, for units with separate radwaste systems, the submittal shall specify the release of radioactive material from each unit.

6.9.4.1 Routine Radioactive Effluent Release Reports covering the operations of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

6.9.4.2 The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the from of an hour-by-hour listing of wind speed, wind direction, atmosphere stability, and precipitation (if measured) on magnetic tape, or in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to individuals due to their activities inside the site boundary (Figures 5-3 and 5-4) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual(ODCM).

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed real individual from reactor releases and other nearby uranium fuel cycle sources including doses from primary effluent pathways and direct radiation for the previous 12 consecutive months to show conformance with 40 CFR 190 "Environmental Radiation Protection Standards for Nuclear Power Operation". Acceptable methods for calculating the dose contributions from Liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The Radioactive Effluent Release Reports shall include the following information for each type of solid waste shipped offsite during the report period:

a. container volume,

b. total curie quantity (specify whether determined by measurement or estimate),

c. principal radionuclides (specify whether determined by measurement or estimate),

d. type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. type of container (e.g., LSA, Type A, Type B, Large Quantity) and

f. solidification agent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a summary of unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.23.2.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records of normal station operation including power levels and periods of operation at each power level.

b. Records of principal maintenance activities, including inspection, repairs, substitution, or replacement of principal items of equipment important to safety.

c. All REPORTABLE EVENTS

d. Records of periodic checks, tests and calibrations.

e. Records of reactor physics tests and other special tests important to safety.

f. Changes to operating procedures important to safety.

g. Records of solid radioactive shipments.

h. Test results, in units of microcuries, for leak tests performed on licensed sealed sources.

i. Results of annual physical inventory verifying accountability of licensed sources on record.

j. Control Room Log Book.

k. Shift Foreman Log Book.

6.10.2 The following records shall be retained for the duration of Operating License DPR-50 unless otherwise specified in 6.10.1 above.

a. Record and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Routine unit radiation surveys and monitoring records.

d. Records of radiation exposure history and radiation exposure status of personnel, including all contractors and unit visitors who enter radioactive material areas.

e. Records of radioactive liquid and gaseous wastes released to the environment, and records of environmental monitoring surveys.

f. Records of transient or operational cycles for those facility components important to safety for a limited number of transients or cycles as defined in the Final Safety Analysis Report.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the Operational Quality Assurance Plan.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of reviews by the Independent Onsite Safety Review Group (previously Plant Operations Review Committee and General Office Review Board minutes).

l. Records of analyses required by the radiological environmental monitoring program.

m. Records of the service lives of all safety related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203 (c)(2) of 10 CFR 20:

a. Each High Radiation Area as defined by paragraph 20.202 (b)(3) shall be barricaded and conspicuously posted as a High Radiation Area, and personnel desiring entrance shall obtain a Radiation Work Permit (RWP). Any individual or group of individuals entering a High Radiation Area shall (a) use a continuously indicating dose rate monitoring device or (b) use a radiation dose rate integrating device which alarms at a pre-set dose level (entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them), or (c) assure that a radiological control technician provides positive control over activities within the area and periodic radiation surveillance with a dose rate monitoring instrument.

b. Any area accessible to personnel where a major portion of the body could receive in any one hour a dose in excess of one thousand mrem shall be locked or guarded to prevent unauthorized entry. The keys to these locked barricades shall be maintained under the administrative control of the respective Radiological Controls Supervisor.

The Radiation Work Permit is not required by Radiological Controls personnel during the performance of their assigned radiation protection duties provided they are following radiological control procedures for entry into High Radiation Areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 GPU Nuclear Corporation initiated changes to the PCP:

1. Shall be submitted to the NRC in the Semiannual Radioactive Effluent Release Report for the period in which the changes were made. This submittal shall contain:

   a. sufficiently detailed information to justify the changes without benefit of additional or supplemental information;

   b. a determination that the changes did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-50

METROPOLITAN EDISON COMPANY
JERSEY CENTRAL POWER AND LIGHT COMPANY
PENNSYLVANIA ELECTRIC COMPANY
GPU NUCLEAR CORPORATION

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

INTRODUCTION

By letter dated May 12, 1986, as supplemented September 11, 1986, GPU Nuclear Corporation, (GPU or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1 (TMI-1). The proposed amendment would revise notification and reporting requirements, as requested by NRC Generic Letter 83-43 dated December 19, 1983, to be consistent with the new requirements in 10 CFR Parts 50.72 and 50.73. In addition, certain administrative changes affecting the same and other pages have been incorporated in this request, which serve to clarify the existing Technical Specifications (TSs) and include additional elements of the Standard Technical Specifications (STS).

EVALUATION

Currently, Administrative Control Specification 6.9.2 "Reportable Occurrences" requires the licensee to report certain types of events either by prompt notification (with written follow-up) or in thirty day written reports. The proposed revisions bring the Three Mile Island, Unit No. 1 Technical Specifications into conformance with new requirements in 10 CFR Parts 50.72 and 50.73. The changes include (1) adding Definition 1/21 "Reportable Events"; (2) deleting unnecessary and conflicting references to reporting requirements in the Limiting Conditions for Operations and Surveillance Requirements sections or otherwise revising previous "Unique Reporting Requirements" into these sections, and (3) revising the Administrative Controls sections to reference 10 CFR Part 50.73 and to delete the previous reporting requirements, now unnecessary or conflicting.

The proposed TS changes also make certain additional revisions to the Technical Specifications as follows:

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1. Deletion of the requirement for certain reports or information deleted from STS (NUREG-0472, Revision 3). This information, although no longer required to be reported, will be available at the site for NRC review.

2. Clarification of TS 6.10.2 to distinguish between the records which are to be retained for the duration of the operating license and those which are required to be retained for at least five years. This change is only to eliminate confusion and does not modify the requirements for record retention.

3. Deletion of TS 6.10.2n concerning the retention of equipment qualification records. These requirements are adequately addressed by regulation in 10 CFR 50.49. Furthermore, this specification referenced TS 6.15 which has been deleted by Amendment No. 102.

4. Deletion of the reference to Regulatory Guide (RG) 10.1 from TS 6.9.1.c. NRC requested distribution of Monthly Operating Reports is not consistent with the guidance specified in RG 10.1, thus this reference is no longer appropriate.

5. Other changes to correct format, grammar, misspellings, other errors from previous amendments, and the addition of language consistent with the Standard Technical Specifications. These changes do not alter any requirements; they serve only to improve the clarity of the specifications.

The staff has evaluated the proposed changes to the TSs and concludes for the reasons stated above that these changes: (1) are in accordance with the changes requested by Generic Letter No. 83-43, (2) are in accordance with the language in the STS, (3) correct existing errors in the current TS,and (4) serve to clarify certain sections. Therefore, these changes are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves changes which are administrative in nature in that they change reporting requirements, delete errors in the existing Technical Specifications, and serve to improve and clarify certain sections. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 14, 1987

Principal Contributors: Walter Baunack, R. Struckmeyer, and M. Miller