April 27, 1989

Docket No. 50-289

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Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 72013, 72014 AND 72081)

The Commission has issued the enclosed Amendment No. 149 to Facility Operating License No. DPR-50 for the Three Mile Island Nuclear Station, Unit No. 1, in response to your letters dated December 2, 1988, December 19, 1988 and January 31, 1989.

The amendment revises the Technical Specifications to (1) change the reactor coolant temperature setpoint at which low temperature overpressure protection is enabled, (2) correct an error resulting from an oversight in License Amendment No. 142, (3) clarify a number of requirements and their bases and (4) provide an option to utilize qualified onsite personnel in certain cases to participate in annual fire protection inspections and audits.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 149 to DPR-50
2. Safety Evaluation

cc w/enclosures:
See next page

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1/1

[TAC NOS. 72013,72014, 72081]

LA:PDI-4	PM:PDI-4	PD:PDI-4	OGC
SNorris	RHernan:cb	JStolz	JM Janice Moore
4/13/89	4/13/89	4/14/89	4/24/89

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CP-1 cc

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

Docket No. 50-289

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and Director - TMI-1
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Division of Reactor Projects I/II
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2. Safety Evaluation

cc w/enclosures:
See next page

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UNITED STATES
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WASHINGTON, D. C. 20555

METROPOLITAN EDISON COMPANY

JERSEY CENTRAL POWER & 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. 149
License No. DPR-50

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

   A. The applications for amendment by GPU Nuclear Corporation, et al. (the licensee) dated December 2 and December 19, 1988 and January 31, 1989, comply 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.

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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 are hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.149 , 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, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 27, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 149

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
ii	ii
viii	viii
3-13a	3-13a
3-14	3-14
3-18c	3-18c
3-18d	3-18d
3-18g	3-18g
3-23	3-23
3-24	3-24
3-29 (Table 3.5-1)	3-29 (Table 3.5-1)
3-37	3-37
3-37a	3-37a
3-61	3-61
3-62a	3-62a
3-106	3-106
3-107	3-107
3-109	3-109
4-4 (Table 4.1-1)	4-4 (Table 4.1-1)
4-8 (Table 4.1-2)	4-8 (Table 4.1-2)
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4-80a	-
4-81	4-81
4-82	4-82
5-1	5-1
5-10	5-10
5-11	-
Figure 5-1	Figure 5-1
Figure 5-2	Figure 5-2
Figure 5-3	Figure 5-3
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Although some leak rates on the order of gallons per minute may be tolerable from a dose point of view, it is recognized that leaks in the order of drops per minute through any of the barriers of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation, and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature and location of the leak, as well as the magnitude of the leakage, must be considered in the safety evaluation.

When reactor coolant leakage occurs to the Reactor Building, it is ultimately conducted to the Reactor Building sump. Although the reactor coolant is safely contained, the gaseous components in it escape to the Reactor Building atmosphere. There, the gaseous components become a potential hazard to plant personnel, during inspection tours within the Reactor Building, and to the general public whenever the Reactor Building atmosphere is periodically purged to the environment.

When reactor coolant leakage occurs to the Auxiliary Building, it is collected in the Auxiliary Building sump. The gases escaping from reactor coolant leakage within the Auxiliary Building will be collected in the Auxiliary and Fuel Handling Building exhaust ventilation system and discharged to the environment via the unit's Auxiliary and Fuel Handling Building vent. Since the majority of this leakage occurs within confined, separately ventilated cubicles within the Auxiliary Building, it incurs very little hazard to plant personnel.

In regard to the surveillance specification 4.2.7, the isolation valves may be tested at a reduced pressure in accordance with the Franklin Research Center Report titled "Primary Coolant System Pressure Isolation Valves for TMI-1" (FRC Task 212) dated October 24, 1980, Section 2.2.2.

When reactor coolant leakage occurs to the nuclear services closed cooling water system, the leakage, both gaseous and liquid, is contained because the nuclear services closed cooling water system surge tank is a closed tank that is maintained above atmospheric pressure. The leakage would be detected by the nuclear services closed cooling water system monitor and by purge tank liquid level, both of which alarm in the control room. Since the nuclear services closed cooling water system's only potential contact with reactor coolant is in the sample coolers, it is considered not to be a hazard. However, if reactor coolant leakage to this receptor occurred and the surge tank's relief valve discharged, radioactive gases could be discharged to the environment via the unit's auxiliary and fuel handling building vent.

Bases (Continued)

When reactor coolant leakage occurs to the intermediate cooling closed cooling water system, the leakage is indicated by both the intermediate cooling water monitor (RM-L9) and the intermediate cooling closed cooling water surge tank liquid level indicator, both of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank.

When reactor coolant leakage occurs to either of the decay heat closed cooling water systems, the leakage is indicated by the affected system's radiation monitor (RM-L2 or RM-L3 for system A and B, respectively) and surge tank liquid level indicator, all four of which alarm in the control room. Reactor coolant leakage to this receptor ultimately could result in radioactive gas leaking to the environment via the unit's auxiliary and fuel handling building vent by way of the atmospheric vent on the surge tank of the affected system.

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the guide lines of 10 CFR 20 would be exceeded. This is shown as follows: if the specific activity of the reactor coolant is $130/\bar{E}$ uCi/ml and the gaseous portion of it (as identified by UFSAR Table 11.1-2) is discharged to the environment via the unit's auxiliary and fuel handling building vent, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $X/Q = 4.5 \times 10^{-6}$ sec/m^3, is 0.34 rem. This may be compared with the 10 CFR 20 guideline of 0.5 rem/year whole body dose.

When the reactor coolant leaks to the secondary sides of either steam generator, all the gaseous components and a very small fraction of the ionic components are carried by the steam to the main condenser. The gaseous components exit the main condenser via the unit's vacuum pump which discharges to the condenser vent past the condenser off-gas monitor. The condenser off-gas monitor will detect any radiation, above background, within the condenser vent.

However, buildup of radioactive solids in the secondary side of a steam generator and the presence of radioactive ions in the condensate can be tolerated to only a small degree. Therefore, the appearance of activity in the condenser off-gas, or any other possible indications of primary to secondary leakage such as water inventories, condensate demineralizer activity, etc., shall be considered positive indication of primary to secondary leakage and steps shall be taken to determine the source and quantity of the leakage.

3.1.12 Pressurizer Power Operated Relief Valve (PORV) and Block Valve

Applicability

Applies to the settings, and conditions for isolation of the PORV.

Objective

To prevent the possibility of inadvertently overpressurizing or depressurizing the Reactor Coolant System.

Specification

3.1.12.1 The PORV shall not be taken out of service, nor shall it be isolated from the system (except that the PORV may be isolated to limit leakage to within the limits of Specification 3.1.6) unless one of the following is in effect:

a. High Pressure Injection Pump breakers are racked out or MU-V16A/B/C/D and MU-V217 are closed.
b. Head of the Reactor Vessel is removed.
c. T_{avg} is above 332°F.

3.1.12.2 The PORV settings shall be as follows, within the tolerances of ± 25 psi and ± 12°F:

Above 275°F - 2450 psig
Below 275°F - 485 psig

3.1.12.3 If the reactor vessel head is installed and Tavg is ≤332°F, High Pressure Injection Pump breakers shall not be racked in unless:

a. MU-V16A/B/C/D and MU-V217 are closed, and
b. Pressurizer level is ≤ 220 inches. If pressurizer level is >220 inches, restore level to ≤ 220 inches within 1 hour.

3.1.12.4 PORV and Block Valve

The PORV and the associated block valve shall be OPERABLE during HOT STANDBY, STARTUP, AND POWER OPERATION:

a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b With the PORV block valve inoperable, within 1 hour either restore the PORV block valve to OPERABLE status or close the PORV (verify closed) and remove power from the PORV; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With either the PORV or block valve inoperable, restore the inoperable valve to operable status prior to startup from the next cold shutdown unless the cold shutdown occurs within 90 Effective Full Power Days (EFPD) of the end of the fuel cycle. If a cold shutdown occurs within this 90-day period, restore the inoperable valve to operable status prior to the startup for the next fuel cycle.

Bases

If the PORV is removed from service, sufficient measures are incorporated to prevent severe overpressurization by either eliminating the high pressure sources or flowpaths or assuring that the RCS is open to atmosphere. In order to prevent exceeding leakage rates specified in T.S. 3.1.6., the PORV may be isolated.

The PORV setpoints are specified with tolerances assumed in the bases for Technical Specification 3.1.2.

With RCS temperatures less than 332°F and the makeup pumps running, the high pressure injection valves are closed and pressurizer level is maintained less than 220 inches to prevent severe overpressurization in the event of any single failure.

Both the PORV and the PORV block valve should be operable during the HOT STANDBY, STARTUP, AND POWER OPERATION. If either the PORV or the PORV block valve are inoperable, the PORV discharge line should be isolated to prevent potential uncontrolled RCS depressurization.

For protection from severe overpressurization during HPI testing, refer to Section 4.5.2.1.c.

Bases

The safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plants ability to deal with large quantities of noncondensible gas which could interfere with natural circulation (i.e., core cooling).

The reactor vessel head vent (RC-V42 & RC-V43 in series) provides the capability of venting noncondensible gases from the majority of the reactor vessel head as well as the Reactor Coolant hot legs (to the elevation of the top of the outlet nozzles) and cold legs (through vessel internals leakage paths, to the elevation of the top of the inlet nozzles). This vent is routed to containment atmosphere.

Venting for the pressurizer steam space (RC-V28 and RC-V44 in series) has been provided to assure that the pressurizer is available for Reactor Coolant System pressure and volume control. This vent is routed to the Reactor Coolant Drain Tank.

Additional venting capability has been provided for the Reactor Coolant hot leg high points (RC-V40A, B, RC-41A, B), which normally cannot be vented through the Reactor vessel head vent or pressurizer steam-space vent. These vents relieve to containment atmosphere through a rupture disk (set at low pressure).

The above vent systems are seismically designed and environmentally qualified in accordance with the May 23, 1980 Commission Order and Memorandum per NUREG-0737, Item II.B.1. The high point vents do not fall within the scope of 10 CFR 50.49, since the vents are not relied upon during or following any design basis event. The power operated valves (2 in series in each flow path) which are powered from emergency buses fail closed on loss of power. All vent valves for the reactor vessel head vent, pressurizer vent and loop B high point vent are powered from the class 1E "B" bus. The vent valves for the loop A high point vent are powered from the class 1E "A" bus. The power operated valves are controlled in the Control Room. The individual vent path lines are sized so that an inadvertent valve opening will not constitute a LOCA as defined in 10 CFR 50.46(c)(1). These design features provide a high degree of assurance that these vent paths will be available when needed, and that inadvertent operation or failures will not significantly hamper the safe operation of the plant.

3.3.3 Exceptions to 3.3.2 shall be as follows:

a. Both core flood tanks shall be operable at all times.

b. Both the motor operated valves associated with the core flood tanks shall be fully opened at all times.

c. One reactor building cooling fan and associated cooling unit shall be permitted to be out-of-service for seven days.

3.3.4 Prior to initiating maintenance on any of the components, the duplicate (redundant) component shall be tested to assure operability.

Bases

The requirements of Specification 3.3.1 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two engineered safeguards makeup pumps, two decay heat removal pumps and two decay heat removal coolers (along with their respective cooling water systems components) are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required because a single core flooding tank has insufficient inventory to reflood the core for hot and cold line breaks.

The operability of the borated water storage tank (BWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA (Reference 2). The limits on BWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain at least one percent subcritical at 70°F without any control rods in the core following mixing of the BWST and RCS water volumes (Reference 3).

The contained water volume limit of 350,000 gallons includes an allowance for water not usable because of tank discharge location. The limits on contained water volume, NaOH concentration and boron concentration ensure a pH value of between 8.5 and 11.0 of the solution sprayed within containment after a design basis accident. The minimum pH of 8.5 assures that iodine will remain in solution while the maximum pH of 11.0 minimizes the potential for caustic damage to mechanical systems and components. Redundant heaters maintain the borated water supply at a temperature greater than 40°F.

The post-accident reactor building emergency cooling may be accomplished by three emergency cooling units, by two spray systems, or by a combination of one emergency cooling unit and one spray system. The specified requirements assure that the required post-accident components are available.

The iodine removal function of the reactor building spray system requires one spray pump and sodium hydroxide tank contents.

The spray system utilizes common suction lines with the decay heat removal system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.2 and 3.3.3 provided requirements in Specification 3.3.4 are met which assure operability of the duplicate components. The specified maintenance times are a maximum. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5.

An allowable maintenance period of up to 72 hours may be utilized if the operability of equipment redundant to that removed from service is demonstrated immediately prior to removal.

In the event that the need for emergency core cooling should occur, operation of one makeup pump, one decay heat removal pump, and both core flood tanks will protect the core. In the event of a reactor coolant system rupture their operation will limit the peak clad temperature to less than 2,300°F and the metal-water reaction to that representing less than 1 percent of the clad.

Two nuclear service river water pumps and two nuclear service closed cycle cooling pumps are required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant.

REFERENCES

(1) Updated FSAR, Section 6.1 - Emergency Core Cooling System

(2) Updated FSAR, Section 14.2.2.3 - Large Break LOCA

(3) Updated FSAR, Section 14.2.2.1 - Fuel Handling Accident

TABLE 3.5-1
INSTRUMENTS OPERATING CONDITIONS

Functional Unit	(A) Minimum Operable Channels	(B) Minimum Degree of Redundancy	(C) Operator Action if Conditions of Column A and B Cannot be Met
A. Reactor Protection System			
1. Manual pushbutton	1	0	Maintain hot shutdown
2. Power range instrument channel	3(a)	1(a)	Maintain hot shutdown
3. Intermediate range instrument channels	1	0	Maintain hot shutdown(b)
4. Source range instrument channels	1	0	Maintain hot shutdown(c)
5. Reactor coolant temperature instrument channels	2	1	Maintain hot shutdown
6. (Deleted)	--	--	- - - -
7. Flux/imbalance/flow	2	1	Maintain hot shutdown
8. Reactor coolant pressure			
a. High reactor coolant pressure instrument channels	2	1	Maintain hot shutdown
b. Low reactor coolant pressure instrument channels	2	1	Maintain hot shutdown

3.5.3 ENGINEERED SAFEGUARDS PROTECTION SYSTEM ACTUATION SETPOINTS

Applicability:

This specification applies to the engineered safeguards protection system actuation setpoints.

Objective:

To provide for automatic initiation of the engineered safeguards protection system in the event of a breach of Reactor Coolant System integrity.

Specification:

3.5.3.1 The engineered safeguards protection system actuation setpoints and permissible bypasses shall be as follows:

Initiating Signal	Function	Setpoint
High Reactor Building Pressure (1)	Reactor Building Spray	≤ 30 psig
	Reactor Building Isolation	≤ 30 psig
	High-Pressure Injection	≤ 4 psig
	Low-Pressure Injection	≤ 4 psig
	Start Reactor Building Cooling & Reactor Building Isolation	≤ 4 psig
Low Reactor Coolant System Pressure	High Pressure Injection	≥ 1600(2) and ≥ 500(3) psig
	Low Pressure Injection	≥ 1600(2) and ≥ 500(3) psig
	Reactor Building Isolation	≥ 1600 psig(2)
4.16 kv E.S. Buses Undervoltage Relays		
Degraded Voltage (5)	Switch to Onsite Power Source and load shedding	3595 volts (4)
Degraded grid timer		10 sec (5)
Loss of voltage	Switch to Onsite Power Source and load shedding	2400 Volts (6)
Loss of voltage timer		1.5 sec (7)

(1) May be bypassed for reactor building leak rate test.

(2) May be bypassed below 1775 psig on decreasing pressure and is automatically reinstated before 1800 psig on increasing pressure.

(3) May be bypassed below 925 psig on decreasing pressure and is automatically reinstated before exceeding 950 psig on increasing pressure.

(4) Minimum allowed setting is 3560 v. Maximum allowed setting is 3650 v.

(5) Minimum allowed time is 8 sec. maximum allowed time is 12 sec.

(6) Minimum allowed setting is 2200 volts, maximum allowed setting is 2860 volts

(7) Minimum allowed time is 1.0 second, maximum allowed time is 2.0 seconds.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a LOCA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressures to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1600 and 500 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum to break sizes and is far enough below normal operating pressure to prevent spurious initiation. Bypass of HPI below 1775 psig and LPI below 925 psig, prevents ECCS actuation during normal system cooldown.

4.16 KV ES Bus Undervoltage Relays

The basis for the degraded grid voltage relay setpoint is to protect the safety related electrical equipment from loss of function in the event of a sustained degraded voltage condition on the offsite power system. The timer setting prevents spurious transfer to the onsite source for transient conditions.

The loss of voltage relay and timers detect loss of offsite power condition and initiate transfer to the onsite source with minimal time delay.

3.15 AIR TREATMENT SYSTEMS

3.15.1 EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the emergency control room air treatment system and its associated filters.

Objective

To specify minimum availability and efficiency for the emergency control room air treatment system and its associated filters.

Specifications

3.15.1.1 Except as specified in Specification 3.15.1.3 below, both emergency treatment systems, AH-E18A fan and associated filter AH-F3A and AH-E18B fan and associated filter AH-F3B shall be operable at all times, per the requirements of Specification 3.15.1.2 below; when containment integrity is required and when irradiated fuel handling operations are in progress.

3.15.1.2 a. The results of the in-place DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show <0.05% DOP penetration and <0.05% halogenated hydrocarbon penetration, except that the DOP test will be conducted with prefilters installed.

b. The results of laboratory carbon sample analysis shall show ≥90% radioactive methyl iodide decontamination efficiency when tested at 125°F, 95% R.H.

c. The fans AH-E18A and B shall each be shown to operate within ± 4000 CFM of design flow (40,000 CFM).

3.15.1.3 From and after the date that one control room air treatment system is made or found to be inoperable for any reason, reactor operation or irradiated fuel handling operations are permissible only during the succeeding 7 days provided the redundant system is demonstrated to be operable per 4.12.1.1 and 4.12.1.3 within 24 hours and daily thereafter.

3.15.1.4 From the date that both control room air treatment systems are made or found to be inoperable or if the inoperable system of 3.15.1.3 cannot be made operable in 7 days, irradiated fuel handling operations shall be terminated in 2 hours and reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 48 hours

3.15.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Applicability

Applies to the reactor building purge air treatment system and its associated filters.

Objective

To specify minimum availability and efficiency for the reactor building purge air treatment system and its associated filters.

Specification

3.15.2.1 Except as specified in Specification 3.15.2.3 below, the Reactor Building Purge Air Treatment System filter AH-F1 shall be operable as defined by the Specification below at all times when containment integrity is required unless the Reactor Building purge isolation valves are closed.

3.15.2.2 a. The results of the in-place DOP and halogenated hydrocarbon tests at maximum available flows on HEPA filters and charcoal adsorber banks for AH-F1 shall show less than 0.05% DOP penetration and less than 0.05% halogenated hydrocarbon penetration; except that the DOP test will be conducted with prefilters installed.

b. The results of laboratory carbon sample analysis for the reactor building purge system filter carbon shall show greater than or equal to 90% radioactive methyl iodide decontamination efficiency when tested at 250°F,95% R.H.

3.15.2.3 From and after the date that the filter AH-F1 in the reactor building purge system is made or found to be inoperable as defined by Specification 3.15.2.2 above, the Reactor Building purge isolation valves shall be closed until the filter is made operable.

Bases

The Reactor Building Purge Exhaust System filter AH-F1 while normally used to filter all reactor building exhaust air. It is necessary to demonstrate operability of these filters to assure readiness for service if required to mitigate a fuel handling accident in the Reactor Building and to assure that 10CFR50 Appendix I limits are met. Reactor Building purging is required to be terminated if the filter is not operable.

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3.22 RADIOACTIVE EFFLUENTS

3.22.1 LIQUID EFFLUENTS

3.22.1.1 CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.22.1.1 The concentration of radioactive material released at anytime from the unit to unrestricted areas (see Figure 5-3) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 3×10^{-3} uCi/cc total activity.

APPLICABILITY: At all times

ACTION:

a. With the concentration of radioactive material released from the unit to unrestricted areas exceeding the above limits, immediately restore concentration within the above limits.

b. If action "a" cannot be met, then be in:

   1. At least HOT STANDBY within 1 hour,
   2. At least HOT SHUTDOWN within the next 6 hours, and
   3. At least COLD SHUTDOWN within the following 30 hours.

BASES

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the unit to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will not result in exposures with (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR Part 20.106 (e) to the population. The concentration limit for noble gases is based upon the assumption the Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

DOSE

LIMITING CONDITION FOR OPERATION

3.22.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from the unit to the site boundary (see Figure 5-3) shall be limited:

a. During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 5 mrem to any organ.

b. During any calendar year to ≤ 3 mrem to the total body and to ≤ 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 10 CFR 20. The dose calculations in the ODCM implement

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 5-3) 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.

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	M	R	
9. Low Reactor Coolant Pressure Channel	S	M	R	
10. Flux-Reactor Coolant Flow Comparator	S	M	R	
11. (Deleted)	--	--	--	
12. Pump Flux Comparator	S	M	R	
13. High Reactor Building Pressure Channel	S	M	R	
14. High Reactor Building Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Tav is greater than 200 F
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Low Pressure Injection Analog Channels	NA	Q	NA	
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or Tav is greater than 200 F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

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TABLE 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2.	Control Rod Movement	Movement of each rod	Every two weeks, when reactor is critical
3.	Pressurizer Safety Valves	Setpoint*	50% each refueling period
4.	Main Steam Safety Valves	Setpoint	25% each refueling period
5.	Refueling System Interlocks	Functional	Start of each refueling period
6.	Main Steam Isolation Valves	(See Section 4.8)	
7.	Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525°F
8.	Deleted		
9.	Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10.	Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation-Visual inspection of Intake Pump House Floor	Each refueling period
		(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11.	Pressurizer Block Valve (RC-V2)	Functional**	Quarterly

* The setpoint of the pressurizer code safety valves shall be in accordance with ASME Boiler and Pressurizer Vessel Code, Section III, Article 9, Winter, 1968.

** Function shall be demonstrated by operating the valve through one complete cycle of full travel.

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4.5 EMERGENCY LOADING SEQUENCE AND POWER TRANSFER, EMERGENCY CORE COOLING SYSTEM & REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.1 Emergency Loading Sequence

Applicability: Applies to periodic testing requirements for safety actuation systems.

Objective: To verify that the emergency loading sequence and automatic power transfer is operable.

Specifications:

4.5.1.1 Sequence and Power Transfer Test

a. During each refueling interval, a test shall be conducted to demonstrate that the emergency loading sequence and power transfer is operable.

b. The test will be considered satisfactory if the following pumps and fans have been successfully started and the following valves have completed their travel on preferred power and transferred to the emergency power as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

-M. U. Pump
-D. H. Pump and D. H. Injection Valves and D. H. Supply Valves
-R. B. Cooling Pump
-R. B. Ventilators
-D. H. Closed Cycle Cooling Pump
-N. S. Closed Cycle Cooling Pump
-D. H. River Cooling Pump
-N. S. River Cooling Pump
-D. H. and N. S. Pump Area Cooling Fan
-Screen House Area Cooling Fan
-Spray Pump. (Initiated in coincidence with a 2 out of 3 R. B. 30 psig Pressure Test Signal.)
-Motor Driven Emergency Feedwater Pump

c. Following successful transfer to the emergency diesel, the diesel generator breaker will be opened to simulate trip of the generator then reclosed to verify block load on the reclosure.

4.5.1.2 Sequence Test

a. At intervals not to exceed 3 months, a test shall be conducted to demonstrate that the emergency loading sequence is operable, this test shall be performed on either preferred power or emergency power.

b. The test will be considered satisfactory if the pumps and fans listed in 4.5.1.1b have been successfully started and the valves listed in 4.5.1.1b have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

Bases

The emergency loading sequence and automatic power transfer controls the operation of the pumps associated with the emergency core cooling system and Reactor Building cooling system.

REFERENCES

(1) Updated FSAR Section 7.0

(2) Updated FSAR Section 1.4

(3) Specification 4.6.1b

4.5.2 EMERGENCY CORE COOLING SYSTEM

Applicability: Applies to periodic testing requirement for emergency core cooling systems.

Objective: To verify that the emergency core cooling systems are operable.

Specification:

4.5.2.1 High Pressure Injection

a. During each refueling interval and following maintenance or modification that affects system flow characteristics, system pumps and system high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable.

   The M. U. Pump and its required supporting auxiliaries will be started manually by the operator and a test signal will be applied to the high pressure injection valves MU-V-16A, B, C, D to demonstrate actuation of the high pressure injection system for emergency core cooling operation.

b. The test will be considered satisfactory if the valves have completed their travel and the M. U. Pumps are running as evidenced by the control board component operating lights. Minimum acceptable injection flow must be greater than or equal to 500 gpm per HPI pump when pump discharge pressure is 600 psig or greater (the pressure between the pump and flow limiting device) and when the RC pressure is equal to or less than 600 psig.

c. Testing which requires HPI flow thru MU-V16A, B, C, D shall be conducted only under either of the following conditions:

   1) T avg shall be greater the 332°F.
   2) Head of the Reactor Vessel shall be removed.

4.5.2.2 Low Pressure Injection

a. During each refueling period and following maintenance or modification that affects system flow characteristics, system pumps and high point vents shall be vented, and a system test shall be conducted to demonstrate that the system is operable. The auxiliaries required for low pressure injection are all included in the emergency loading sequence specified in 4.5.1.

b. The test will be considered satisfactory if the decay heat pumps listed in 4.5.1.1b have been successfully started and the decay heat injection valves and the decay heat supply valves have completed their travel as evidenced by the control board component operating lights. Flow shall be verified to be equal or greater than the flow assumed in the Safety Analysis for the single corresponding RCS pressure used in the test.

c. When the Decay Heat System is required to be operable, the correct position of DH-V-19A/B shall be verified by observation within four hours of each valve stroking operation or valve maintenance, which effects the position indicator.

4.5.2.3 Core Flooding

a. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. During depressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core cooling flooding tank discharge lines operate properly.

b. The test will be considered satisfactory if control board indication of core flooding tank level verifies that all valves have opened.

4.5.2.4 Component Tests

a. At intervals not to exceed 3 months, the components required for emergency core cooling will be tested.

b. The test will be considered satisfactory if the pumps and fans have been successfully started and the valves have completed their travel as evidenced by the control board component operating lights, and either the station computer or pressure/flow indication.

Bases

The emergency core cooling systems are the principal reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storgage tank through each of the injection lines and back to the tank.

The minimum acceptable HPI/LPI flow assures proper flow and flow split between injection legs.

With the reactor shutdown, the valves in each core flooding lines are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

b. Reactor Building Cooling and Isolation Systems

1. During each refueling period, a system test shall be conducted to demonstrate proper operation of the system. A test signal will actuate the R.B. emergency cooling system valves to demonstrate operability of the coolers.

2. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights, and either the station computer or local verification.

4.5.3.2 Component Tests

a. At intervals not to exceed three months, the components required for reactor building cooling and isolation will be tested.

b. The test will be considered satisfactory if the valves have completed their expected travel as evidenced by the control board component operating lights, and either the station computer or local verification.

Bases

The reactor building cooling and isolation systems and reactor building spray system are designed to remove the heat in the containment atmosphere to prevent the building pressure from exceeding the design pressure.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by the operator action. With the reactor building spray inlet valves closed, low pressure air can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves and instrumentation of the reactor building cooling system are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the reactor building during power operations to inspect and maintain this equipment.

The reactor building fans are normally operating periodically, constituting the test that these fans are operable.

Reference

(1) FSAR, Section 6

4.6 EMERGENCY POWER SYSTEM PERIODIC TESTS

Applicability: Applies to periodic testing and surveillance requirement of the emergency power system.

Objective: To verify that the emergency power system will respond promptly and properly when required.

Specification:

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

a. Manually-initiate start of the diesel generator, followed by manual synchronization with other power sources and assumption of load by the diesel generator up to the name-plate rating (3000 kw). This test will be conducted every month on each diesel generator. Normal plant operation will not be effected.

b. Automatically start and loading the emergency diesel generator in accordance with Specification 4.5.1.1.b/c including the following. This test will be conducted every refueling interval on each diesel generator.
   (1) Verify that the diesel generator starts from ambient condition upon receipt of the ES signal and is ready to load in $\leq$10 seconds.

   (2) Verfiy that the diesel block loads upon simulated loss of offsite power in $\leq$30 seconds.

   (3) The diesel operates with the permanently connected and auto connected load for $\geq$5 minutes.

   (4) The diesel engine does not trip when the generator breaker is opened while carrying emergency loads.

   (5) The diesel generator block loads and operates for $\geq$5 minutes upon reclosure of the diesel generator breaker.

c. Each diesel generator shall be given an inspection at least annually in accordance with the manufacturer's recommendations for this class of stand-by service.

4.6.2 Station Batteries

a. The voltage, specific gravity, and liquid level of each cell will be measured and recorded monthly.

b. The voltage and specific gravity of a pilot cell will be measured and recorded weekly.

c. Each time data is recorded, new data shall be compared with old to detect signs of abuse or deterioration.

4.12 AIR TREATMENT SYSTEM

4.12.1 EMERGENCY CONTROL ROOM AIR TREATMENT SYSTEM

Applicability

Applies to the emergency control room air treatment system and associated components.

Objective

To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.1.1 At least every refueling interval or once every 18 months, whichever comes first, the pressure drop across the combined HEPA filters and charcoal adsorber banks of AH-F3A and 3B shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%).

4.12.1.2 a. The tests and sample analysis required by Specification 3.15.1.2 shall be performed initially and at least once per year for standby service or after every 720 hours of system operation and following significant painting, steam, fire or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing which could affect the HEPA filter bank bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing which could effect the charcoal adsorber bank bypass leakage.

d. Each AH-E18A and B (AH-F3A and B) fan/filter circuit shall be operating at least 10 hours every month.

4.12.1.3 At least once per refueling interval or once every 18 months, whichever comes first, automatic initiation of the Control Building isolation and recirculation Dampers AH-D28, 37, 39, and 36 shall be demonstrated as operable.

4.12.1.4 An air distribution test shall be performed on the HEPA filter bank initially, and after any maintenance or testing that could affect the air distribution within the system. The air distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 40,000 cfm (±10%) flow rate.

4.12.2 REACTOR BUILDING PURGE AIR TREATMENT SYSTEM

Applicability: Applies to the reactor building purge air treatment system and associated components.

Objective: To verify that this system and associated components will be able to perform its design functions.

Specification

4.12.2.1 At least once per refueling interval or once per 18 months, whichever comes first it shall be demonstrated that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at system design flow rate (±10%).

4.12.2.2 a. The tests and sample analysis required by Specification 3.15.2.2, shall be performed initially, once per refueling interval or 2 years, whichever comes first, or within 30 days prior to the movement of irradiated fuel in containment and following significant painting, steam, fire, or chemical release in any ventilation zone communicating with the system that could contaminate the HEPA filters or charcoal adsorbers.

b. DOP testing shall be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing which could affect HEPA frame bypass leakage.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing which could affect the charcoal adsorber bank bypass leakage.

d. The DOP and halogenated hydrocarbon testing shall be performed at the maximum available flow considering physical restrictions, i.e., purge valve position, and gaseous radioactive release criteria.

e. Each refueling, AH-E7A&B shall be shown to operate within ± 5000 cfm of design flow (50,000 cfm) with purge valves fully open.

4.12.2.3 An air distribution test shall be performed on the HEPA filter bank initially and after any maintenance or testing that could affect the air distribution within the system. The air distribution across the HEPA filter bank shall be uniform within ±20%. The test shall be performed at 50,000 cfm (±10%) flow rate with purge valves fully open.

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once every refueling interval to show system performance capability.

Amendment No. ~~55~~, ~~68~~, ~~108~~, 149

4.15 MAIN STEAM SYSTEM INSERVICE INSPECTION

Applicability

This technical specification applies to the inservice inspection of those welds in the main steam system identified as Numbers 3, 4, and 5 of Figure 6, Supplement 2, Part IX and Number 3 of Figure 9, Supplement 2, Part IX.

Objective

The objective of this inservice inspection program is to provide assurance of the continuing integrity of that portion of the main steam system in which a postulated failure would produce pressures in excess of the compartment wall and/or slab capacities.

Specification

4.15.1. The four weld joints identified above shall be 100 percent ultrasonically inspected in accordance with the ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems dated January 1, 1970 as modified by the Winter 1970 Addenda and the provisions of Appendix IX-3400 of Section III of the ASME Boiler and Pressure Vessel Code. Inspections are to be performed prior to startup and subsequently at 3-1/2 year intervals (or nearest refueling outage).

Prior to initial plant operation a preoperational inspection of the identified weld joints will be performed and any data acquired will be recorded to form a baseline on which to compare results of subsequent inspections.

Upon completion of several inspection cycles, the technical benefit of the inspection program frequency will be reviewed. The conclusions of this review shall be submitted to the NRC for evaluation.

Bases

Calculations reveal that postulated breaks in the main steam lines at the containment penetrations in small compartments No. 2 and No. 5 could produce pressures in excess of wall and/or slab capacities.

4.16 REACTOR INTERNALS VENT VALVES SURVEILLANCE

Applicability

Applies to Reactor Internals Vent Valves.

Objective

To verify that no reactor internals vent valve is stuck in the open position and that each valve continues to exhibit freedom of movement.

Specification

	Item	Test	Frequency
4.16.1	Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 400 lbs. (applied vertically upward).	Each Refueling Shutdown

Bases

Verifying vent valve freedom of movement insures that coolant flow does not bypass the core through reactor internals vent valves during operation and therefore insures the conservatism of Core Protection Safety limits as delineated in Figures 2.1-1 and 2.1-3, and the flux/flow trip setpoint.

c. Refueling Outage Inspections

At least once each refueling cycle during shutdown, a visual inspection shall be performed of all safety related snubbers attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify: (1) that there are no visible indications of damage or impaired operability and (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible, and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.17-1f. When the reservoir outlet port of a snubber is found to be uncovered by fluid, the snubber shall only be declared operable if functional testing in both extension and retraction directions is satisfactory and an engineering evaluation concludes that this snubber is operable.

e. Functional Tests*

At least once each refueling interval during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected prior to the test period, or the sample plan used in the prior test period shall be used:

1) At least 10% of the total each type of snubber in use in the plant shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.17.1f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

* The four 550,000 lb reactor coolant pump snubbers are not included. The functional test program for reactor coolant pump snubbers is implemented in accordance with the schedule and other requirements of the snubber testing program.

SURVEILLANCE REQUIREMENTS (Continued)

i. Snubber Seal Service Program

A snubber seal service life program shall be developed whereby the seal service life of hydraulic snubbers is monitored to ensure that the service life is not exceeded between surveillance inspections. The designated service life for the various seals shall be established based on engineering information. The seals shall be replaced so that the indicated service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.m.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss of coolant accident requiring actuation of the engineering safeguards, or

4. A major main steam line or feedwater line break.

4.19.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawing or specifications. Eddy current testing indications below 20% of the nominal tube wal thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

6. Repair Limit means the extent of degradation at or beyond which the tube shall be repaired or removed from service because it may become unserviceable prior to the next inspection.

   This limit is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steam line or feedwater line break as specified in 4.19.3.c., above.

8. Tube Inspection means an inspection of the steam generator tube from the bottom of the upper tubesheet completely to the top of the lower tubesheet, except as permitted by 4.19.2.b.2, above.

4.19.4 Acceptance Criteria (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, 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 3 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.

The program for inservice inspection of steam generator tubes is based on modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Bases (Continued)

The Unit is expected to be operated in a manner such that the primary and secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the primary or secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result.

The extent of steam generator tube leakage due to cracking would be limited by the secondary coolant activity, Specification 3.1.6.3.

The extent of cracking during plant operation would be limited by the limitation of total steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gpm). Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and repaired or removed from service.

Wastage-type defects are unlikely with proper chemistry treatment of the primary or the secondary coolant. However, even if a defect would develop in service, it will be found during scheduled in-service steam generator tube examinations. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Removal from service by plugging, or repair by kinetic expansion, sleeving, or other methods, will be required for degradation equal to or in excess of 40% of the tube nominal wall thickness.

Where experience in similar plants with similar water chemistry, as documented by USNRC Bulletins/Notices, indicate critical areas to be inspected, at least 50% of the tubes inspected should be from these critical areas. First sample inspections sample size may be modified subject to NRC review and approval.

5.0 DESIGN FEATURES

5.1 SITE

Applicability

Applies to the location and extent of the exclusion boundary, restricted area, and low population zone.

Objective

To define the above by location and distance description.

Specification

5.1.1 The Three Mile Island Nuclear Station Unit 1 is located in an area of low population density about ten miles southeast of Harrisburg, PA. It is in Londonderry Township of Dauphin County, Pennsylvania, about two and one-half miles north of the southern tip of Dauphin County, where Dauphin is coterminal with York and Lancaster Counties. The station is located on an island approximately three miles in length situated in the Susquehanna River upstream from York Haven Dam. Figure 5-1 is an extended plot plan of the site showing the plant orientation and immediate surroundings. The Exclusion Area as defined in 10 CFR 100.3, is a 2,000 ft. radius, including portions of Three Mile Island, the river surface around it, and a portion of Shelley Island, which is owned by Met Ed. The minimum distance of 2,000 ft. occurs on the shore of the mainland in a due easterly direction from the plant as shown on Figure 5-1 for the Exclusion Area. Figure 5-3 showing the physical location of the fence defines the "Restricted Area" surrounding the plant. The minimum distance of the "Restricted Area" is approximately 560 feet and is from the centerline of the TMI Unit 2 Reactor Building to a point on the westerly shoreline of Three Mile Island. The minimum distance to the outer boundary of the low population zone is two miles as shown on T.S. Figure 5-2, which also depicts the site topography for a radius of five miles. T.S. Figure 5-3 depicts the locations of gaseous effluent release points and liquid effluent outfalls (as tabularized on page 5-10), and the meteorological tower location (designated as 'weather tower' on the figure).

ELEVATIONS FOR GASEOUS EFFLUENT RELEASE POINTS
(See Figure 5-3)

Unit 1 Stack	483' 7"
Unit 1 Turbine Building	425' 4"
Unit 1 Fuel Handling Building ESF Vent Stack	348'

LOCATIONS OF LIQUID EFFLUENT OUTFALLS PURSUANT TO NPDES
(See Figure 5-3)

Outfall No.	Description
DSN 001	Main Station Discharge
DSN 002	Emergency Discharge from Unit 2 (if DSN 001 is blocked)
DSN 003	Emergency Discharge from Unit 1 (if DSN 001 is blocked)
DSN 004	Emergency Discharge from Unit 1 (if Unit 1 NDCT blocked)
DSN 005	Stormwater and yard drainage and dewatering of natural draft cooling towers, maintenance dredging desiltation and basin dewatering, fire brigade training facility runoff, fire service water runoff.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION(iii)

LICENSE CATEGORY QUALIFICATIONS	Tave > 200°	Tave ≤ 200°
SRO(iv)	2	1(i)
RO(iv)	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1(ii)	None Required

(i) Does not include the Licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising (a) irradiated fuel handling and transfer activities onsite, and (b) all unirradiated fuel handling and transfer activities to and from the Reactor Vessel.

(ii) May be on a different shift rotation than licensed personnel.

(iii) Except for the Shift Supervisor, shift crew composition may be one less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an incoming shift crewman being late or absent.

(iv) Pursuant to the requirements of 10 CFR 50.54(m).

6.5.1.9 The Emergency Plan and implementing procedures shall be reviewed by a knowledgeable individual(s)/group other than the individual(s)/group which prepared them.

6.5.1.10 A knowledgeable individual(s)/group shall review every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports to the Vice President TMI-1 covering evaluations, recommendations and disposition of the corrective action to prevent recurrence.

6.5.1.11 Major changes to radwaste systems shall be reviewed by a knowledgeable individual(s)/group other than the individuals(s)/group which prepared them.

6.5.1.12 Individuals responsible for reviews performed in accordance with 6.5.1.1 through 6.5.1.4 shall include a determination of whether or not additional cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the appropriate personnel. Individuals responsible for reviews considered under 6.5.1.1 through 6.5.1.5 shall render determinations in writing with regard to whether or not 6.5.1.1 through 6.5.1.5 constitute an unreviewed safety question.

RECORDS

6.5.1.13 Written records of activities performed under Specifications 6.5.1.1 through 6.5.1.11 shall be maintained.

QUALIFICATIONS

6.5.1.14 Responsible Technical Reviewers shall meet or exceed the qualifications of ANSI/ANS 3.1 of 1978 Section 4.6, or 4.4 for applicable disciplines, or have 7 years of appropriate experience in the field of his specialty. Credit toward experience will be given for advanced degrees on a one-to-one basis up to a maximum of two years. Responsible Technical Reviewers shall be designated in writing.

6.5.2 INDEPENDENT SAFETY REVIEW FUNCTION

6.5.2.1 The Vice President of each division within GPU Nuclear Corporation shall be responsible for ensuring the independent safety review of the subjects described in 6.5.2.5 within his assigned area of safety review responsibility, as assigned in the GPUN Review and Approval Matrix.

6.5.2.2 Independent safety review shall be completed by an individual/group not having direct responsibility for the performance of the activities under review, but who may be from the same functionally cognizant organization as the individual/group performing the original work.

6.5.2.3 GPU Nuclear Corporation shall collectively have or have access to the experience and competence required to independently review subjects in the following areas:

dures at least once per 24 months.

i. The Process Control Program and implementing procedures for solidification of radioactive wastes at least once per 24 months.

j. The performance of activities required by the Quality Assurance Program to meet criteria of Regulatory Guide 4.15, December, 1977 at least once per 12 months.

k. Any other area of unit operation considered appropriate by the IOSRG or the Office of the President - GPUNC.

6.5.3.2 Audits of the following shall be performed under the cognizance of the vice president responsible for technical support:

a. An independent fire protection and loss prevention program inspection and audit shall be performed annually utilizing either qualified licensee personnel or an outside fire protection firm.

b. An inspection and audit of the fire protection and loss prevention program, by an outside qualified fire consultant at intervals no greater than 3 years.

RECORDS

6.5.3.3 Audit reports encompassed by sections 6.5.3.1 and 6.5.3.2 shall be forwarded for action to the management positions responsible for the areas audited within 60 days after completion of the audit. Upper management shall be informed per the Operation Quality Assurance Plan.

6.5.4 INDEPENDENT ONSITE SAFETY REVIEW GROUP (IOSRG) STRUCTURE

6.5.4.1 The IOSRG shall be a full-time group of engineers, experienced in nuclear power plant engineering, operations and/or technology, independent of the unit staff, and located on site.

ORGANIZATION

6.5.4.2 a. The IOSRG shall consist of a manager and a minimum staff of 3 members who meet the qualifications of 6.5.4.5. Group expertise shall be multi-disciplined.

b. In the event of an unanticipated vacancy in the IOSRG staff, the number of staff can be two (2) members for a period of not to exceed six (6) months while the vacancy is being filled.

c. The IOSRG shall report to the director responsible for nuclear safety assessment.

6.9.4.2.5 The Radioactive Effluent Release Reports shall include the instrumentation not returned to OPERABLE status within 30 days per TS 3.21.1.b and TS 3.21.2.b.

6.9.4.3 The following information shall be included in the Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year.

6.9.4.3.1 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 form 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.

6.9.4.3.2 The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year 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.

6.9.4.3.3 The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the site boundary (Figure 5-3) 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).

6.9.4.3.4 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.

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 related to nuclear safety.

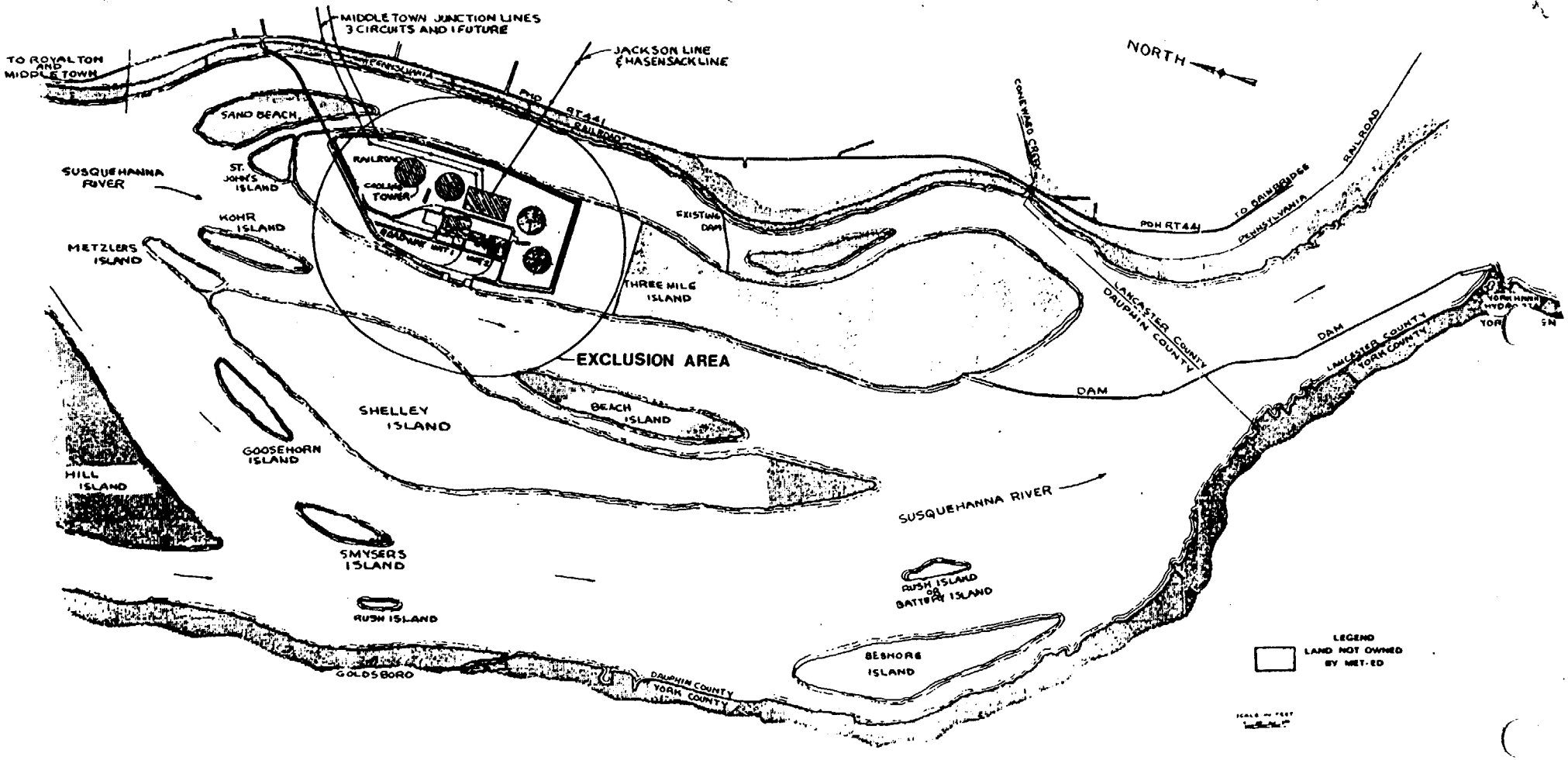
c. All REPORTABLE EVENTS

d. Records of periodic checks, tests and calibrations.

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

f. Changes to procedures required by Specification 6.8.1.

g. Records of solid radioactive shipments.



Amendment No. 149

GPU Nuclear

Extended Plot Plan

Three Mile Island Nuclear Station

Fig. 5-1

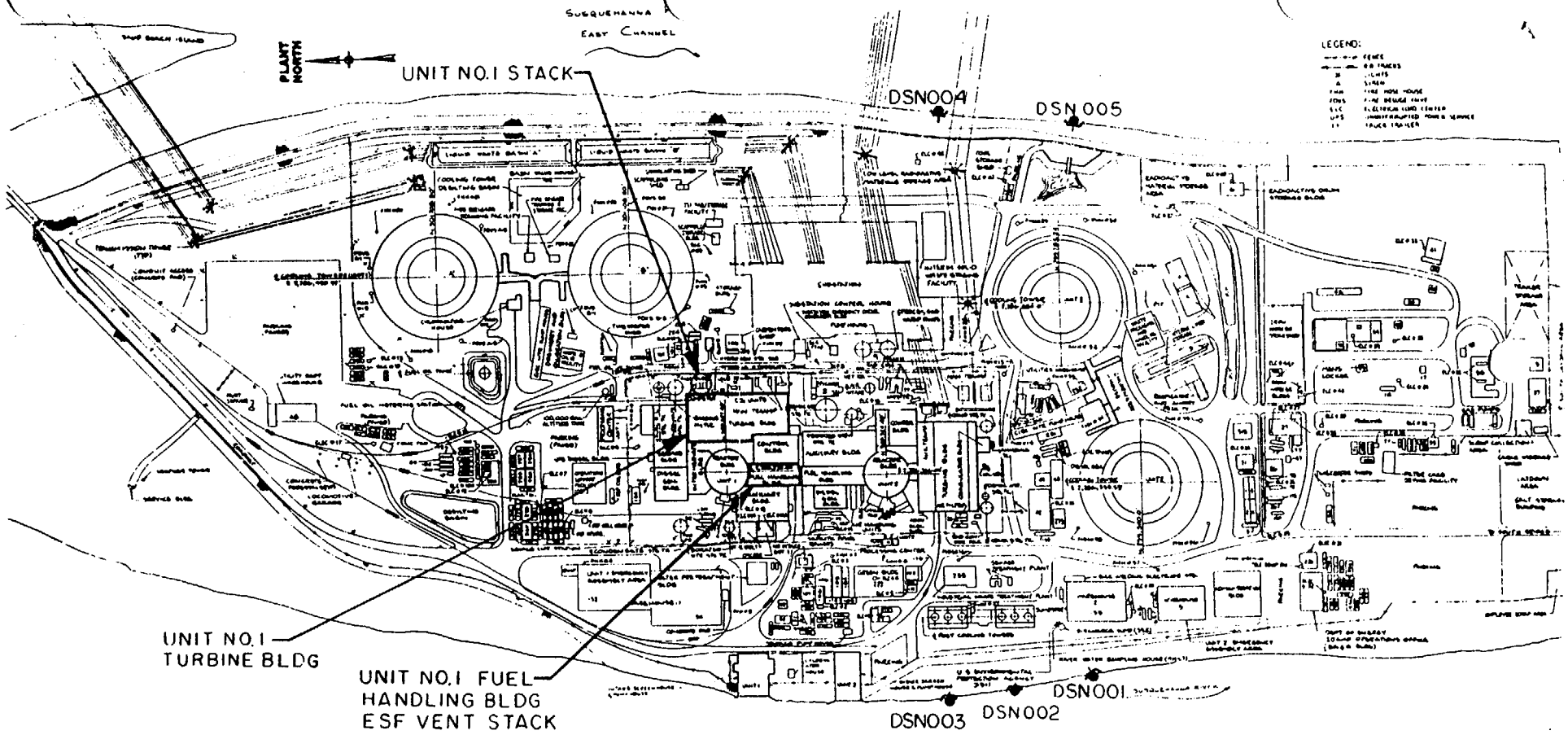
CONTOUR INTERVAL 20 FEET
DATUM IS MEAN SEA LEVEL

GPU Nuclear

Site Topography
5 Mile Radius

Three Mile Island Nuclear Station

Fig. 5-2



GPU Nuclear

Gaseous Effluent Release Points and Liquid Effluent Outfall Locations

Three Mile Island Nuclear Station

Fig. 5

Amendment No. 149

IE-120-CI-001, REV. 9

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 149 TO FACILITY OPERATING LICENSE NO. DPR-50

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

THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-289

INTRODUCTION

GPU Nuclear Corporation (GPU) submitted Technical Specification Change Request (TSCR) Nos. 188, 186 and 187 on December 2, 1988, December 19, 1988 and January 31, 1989, respectively. All three TSCR's involve changes to the Technical Specifications (TS) or their bases that are primarily administrative or editorial in nature: TSCR No. 186 also has a technical aspect to it that has been reviewed by the NRC staff as discussed below. Because of the similar nature of these three requests the staff has elected to issue one overall license amendment in the interest of efficiency.

EVALUATION

TSCR No. 186

This amendment application proposed raising the TS temperature value at which low temperature overpressure protection becomes mandatory from 320°F to 332°F. Presenently TS Sections 3.1.12 and 4.5.2.1.c require specific administrative measures to disable the ability of the High Pressure Injection (HPI) system to overpressurize the Reactor Coolant System (RCS) when RCS average temperature is below 320°F. These two TS sections deal with power operated relief valve (PORV) isolation and HPI flow verification testing, respectively. Raising the temperature value by 12°F to 332°F adds a slight conservatism to the TS since these actions would occur 12°F earlier in a cooldown and the phenomenon of concern is overpressurization when the plant is at low temperatures. Plant operating procedures for heatup and cooldown currently contain this administrative control and specify 332°F as the temperature at which the overpressure protection is to be provided. An additional minor change was added to section 3.1.12.3 that allows a 1 hour period to restore pressurizer level to less than 220 inches in the event it exceeds this limit.

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P PDC

GPU performed a technical evaluation in December 1987 to provide a basis to conclude that the current TMI-1 pressure vs. temperature operating curves (approved by License Amendment No. 134 dated November 18, 1987) do not impact nuclear safety nor pose any unresolved safety issues. This evaluation included an examination of low temperature overpressurization issues including potential initiating events. One of those events is erroneous actuation of the HPI system or opening of an HPI or makeup valve with a makeup pump operating with the plant at low temperature. The specific valves involved are HPI isolation valves mu-v-16A,-16B,-16C,-16D and high capacity makeup valve mu-v-217. Administrative controls in the plant heatup and cooldown procedures require shutting these valves and tagging open their power supply circuit breakers at a specified RCS temperature. The safety evaluation postulated a scenario where the PORV could fail to open on an overpressure event (with the plant shut down) allowing pressure to rise to the code safety relief setpoint of 2750 psig. To provide more conservative protection against low temperature overpressurization during this scenario, the heatup and cooldown procedures were revised to require shutting and disabling these valves at 332°F rather than at 320°F. Plant operating procedures and administrative controls are allowed to be more conservative than the TS, such as in this case. However, GPU has proposed to change the TS in a conservative direction to be consistent with the revised plant procedures.

The staff has reviewed TSCR No. 186, the November 1987 GPU technical evaluation discussed above, the plant heatup and cooldown procedures and the plant temperature/pressure curves and has concluded that this TS change is appropriate. Raising the temperature at which the subject administrative controls (isolating HPI from the RCS) are executed represents a conservatism and eliminates a possible source of operator confusion by making the TS consistent with the operating procedures. Adding the 1 hour action statement to TS section 3.1.12.3 clarifies actions required if the pressurizer level limit is exceeded and is consistent with the standard TS.

TSCR NO. 187

The changes requested by this application are a compilation of editorial changes for clarification of the TS and to correct existing errors, eliminate obsolete terms such as "AEC", upgrade the bases sections and eliminate unnecessary footnotes. A detailed discussion of each change is presented in the GPU change request dated January 31, 1989 and is not repeated here. However, each change was reviewed by the staff for validity and appropriateness. Only those changes to the TS (not including "bases" sections) having any technical significance are discussed in this evaluation.

The NRC issued License Amendment No. 142 in July 1987 in support of Cycle 7 of operation. Based on justification provided by the licensee, the requirement for a variable low RCS pressure reactor trip was eliminated. However, the reference to this instrument on Table 3.5-1 and the surveillance for this trip on Table 4.1-1 were inadvertently not omitted from the TS. This amendment corrects that error.

A number of footnotes throughout TS Sections 3 and 4 contained requirements to be met prior to or during Cycle 5 of operation. The plant is now in Cycle 7 and removal of these footnotes is therefore appropriate.

Section 4.19 of the TS, "OTSG Tube Inservice Inspection", presently contain separate repair limits for primary side defects which are stated to apply only until the 6R Refueling Outage. That outage occurred in 1986-87 and it is appropriate to remove these limits from the TS.

Clarification was proposed regarding OTSG tube repair methods that might be used to meet the operability criteria of TS 4.19.4.b. These criteria, as written, could be misconstrued to exclude all repair methods other than kinetic expansion. The staff concurs with this clarification.

The information concerning liquid effluent outfall descriptions has been upgraded to be consistent with the current NPDES (see page 5-10).

TSCR NO. 188

This application proposes removal of the word "offsite" in TS administrative section 6.5.3.2 dealing with conduct of an annual fire protection audit. The licensee considers that sufficient procedural controls exist to assure that these audits are objective and independent without having the TS require conduct by offsite personnel. This request is consistent with guidance provided by the NRC in Generic Letter No. 82-21 which states "the annual item i audits may be performed by qualified utility personnel who are not directly responsible for the site fire protection program or by an outside independent fire protection consultant."

The staff finds the proposed change acceptable in that it is consistent with NRC guidance.

ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and makes editorial corrections. We have determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released off site, and that there is no significant increase in individual or cumulative occupational exposure. The staff has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (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 (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: R. Hernan

Dated: April 27, 1989