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Docket No. 50-289

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P. O. Box 480
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Dear Mr. Hukill:

SUBJECT: AMENDMENT NO.108 TO FACILITY OPERATING LICENSE NO. DPR-50

The Commission has issued the enclosed Amendment No.108 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 TS Change Request No. 116, Rev. 1, dated November 24, 1983, and supplements dated June 5 and December 3, 1984.

This amendment revises the limits on primary coolant activity and incorporates additional restrictions on vent/purge valve operability and surveillance. This amendment also revises the reactor building purge air system TSs and surveillance requirements for containment area monitors and fire hose stations. These changes make the respective TSs compatible with the purge/vent valve requirements and containment accessibility limits. Finally, the hydrogen purge system TSs are removed because this system is no longer needed to perform a safety-related function.

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

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 108
2. Safety Evaluation

cc w/enclosures:
See next page

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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. 108
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 November 24, 1983, as revised and supplemented June 5, 1984, and December 3, 1984, 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.

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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 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 108, 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, and revised procedures are to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 8, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-50

DOCKET NO. 50-289

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

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3.1.4 REACTOR COOLANT SYSTEM ACTIVITY

3.1.4.1 LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and
b. Less than or equal to 100/$\bar{E}$ microcuries/gram.*

3.1.4.2 APPLICABILITY: at all times except refueling.

3.1.4.3 ACTION:

MODES: Power Operation, Start-up, Hot Standby

a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.1-2a, operation may continue for up to 48 hours** provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12 month period during any fuel cycle. With the total cumulative operating time at a primary coolant specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period during any fuel cycle, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 within 30 days indicating the number of hours of operation above this limit.

b. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours** during one continuous time interval or exceeding the limit line shown on Figure 3.1-2a, be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when DOSE EQUIVALENT I-131 is below 1.0 microcuries/gram.

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

MODES: at all times except refueling.

d. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries/gram perform the sampling and analysis requirements of Table 4.1-3 until the specific activity of the primary coolant is restored to within its limits. A Report shall be prepared and submitted to the Commission within 30 days. This report shall contain the results of the specific activity analyses together with the following information:

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

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

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

2. Fuel burnup by core region,

3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,

4. History of de-gassing operations, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and

5. The time duration when the specific activity of the primary coolant exceeded 1.0 microcurie/gram DOSE EQUIVALENT I-131.

BASES

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will be well within the Part 100 limit following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative, in that the specific site parameters of TMI-1, such as site boundary, location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.1-2a, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.1.2a must be restricted to no more than 800 hours per year (approximately 10 percent of the units yearly operating time) since the activity levels allowed by Figure 3.1-2a increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. Reporting any cumulative operating time over 500 hours in any 6 consecutive month period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will alert the NRC to the situation and allow sufficient time for evaluation and appropriate action before reaching the 800 hour limit.

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

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

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

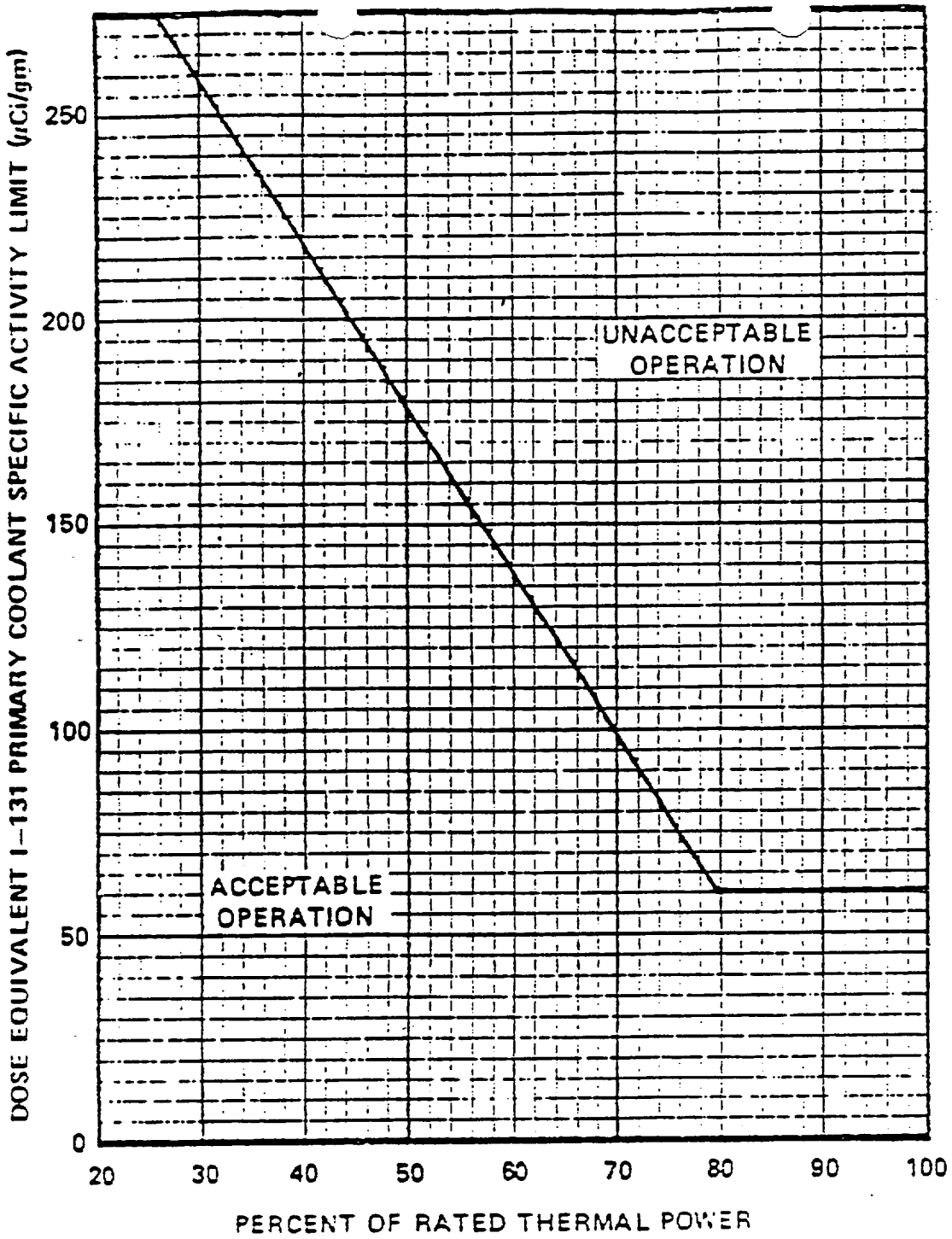

FIGURE 3.1-2a

Dose equivalent I-131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131.

3.1.5 CHEMISTRY

Applicability

Applies to acceptable concentrations of impurities for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities.

Specification

3.1.5.1 If the concentration of oxygen in the primary coolant exceeds 0.1 ppm during power operation, corrective action shall be initiated within eight hours to return oxygen levels to $\leq$ 0.1 ppm.

3.1.5.2 If the concentration of chloride in the primary coolant exceeds 0.15 ppm during power operation, corrective action shall be initiated within eight hours to return chloride levels to $\leq$ 0.15 ppm.

3.1.5.3 If the concentration of fluorides in the primary coolant exceeds 0.10 ppm following modifications or repair to the primary system involving welding, corrective action shall be initiated within eight hours to return fluoride levels to $\leq$ 0.10 ppm.

3.1.5.4 If the concentration limits for oxygen, chloride or fluoride given in 3.1.5.1, 3.1.5.2, and 3.1.5.3 above are not restored within 24 hours of detection, the reactor shall be placed in a hot shutdown condition within 12 hours thereafter. If the normal operational limits are not restored within an additional 24-hour period, the reactor shall be placed in a cold shutdown condition within 24 hours thereafter.

3.1.5.5 If the oxygen, chloride, or fluoride concentration of the primary coolant system exceeds 1.0 ppm the reactor shall be brought to the hot shutdown condition using normal shutdown procedure and action is to be taken to return the system to within normal operation specifications. If normal operating specifications have not been reached in 12 hours, the reactor will then be brought to a cold shutdown condition.

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack.[1,2]

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical. The requirement that the oxygen concentration not exceed 0.1 ppm during power operation is added assurance that stress corrosion cracks will not occur.[4]

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchange resin, or increase the hydrogen concentration in the makeup tank).

3.6 REACTOR BUILDING

Applicability

Applies to the containment integrity of the reactor building as specified below.

Objective

To assure containment integrity.

Specification

3.6.1 Containment integrity as defined in Section 1.7, shall be maintained whenever all three of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200° F or greater.

c. Nuclear fuel is in the core.

3.6.2 Containment integrity shall be maintained when both the reactor coolant system is open to the containment atmosphere and a shutdown margin exists that is less than that for a refueling shutdown.

3.6.3 Positive reactivity insertions which would result in a reduction in shutdown margin to less than 1% $\Delta k/k$ shall not be made by control rod motion or boron dilution unless containment integrity is being maintained.

3.6.4 The reactor shall not be critical when the reactor building internal pressure exceeds 2.0 psig or 1.0 psi vacuum.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and are conspicuously marked.

3.6.6 While the reactor is critical, if a reactor building isolation valve (other than a purge valve) is determined to be inoperable in a position other than the required position, the other reactor building isolation valve in the line shall be tested to insure OPERABILITY. If the inoperable valve is not restored within 48 hours, the OPERABLE valve will be closed or the reactor shall be brought to HOT SHUTDOWN within the next 6 hours and to the COLD SHUTDOWN condition within an additional 30 hours.

3.6.7 The hydrogen recombiner shall be operable during REACTOR CRITICAL, HOT STANDBY and POWER OPERATION. With the hydrogen recombiner inoperable, restore the recombiner to operable status or bring the reactor to HOT SHUTDOWN within seven (7) days.

3.6.8 While containment integrity is required (See TS 3.6.1), if a 48" reactor building purge valve is found to be inoperable perform either 3.6.8.1 or 3.6.8.2 below.

3.6.8.1 If inoperability is due to reasons other than excessive combined leakage close the associated valve and within 24 hours verify that the associated valve is OPERABLE. Maintain the associated valve closed until the faulty valve can be declared operable. If neither purge valve in the penetration can be declared OPERABLE within 24 hours, be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3.6.8.2 If inoperability is due to excessive combined leakage (See T.S. 4.4.1.7.1), within 48 hours restore the leaking valve to OPERABILITY or perform either a or b below.

a. Manually close both associated reactor building isolation valves and meet the leakage criteria of T.S. 4.4.1.7.1 and perform either (1) or (2) below.

(1) restore the leaking valve to OPERABILITY within the following 72 hours.

(2) maintain both valves closed by administrative controls, verify both valves are closed at least once per 31 days and perform the interspace pressurization test of T.S. 4.4.1.7.1 every 3 months. In order to accomplish repairs one containment purge valve may be opened for up to 72 hours following successful completion of an interspace pressurization test.

b. Be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.6.9 Except as specified in 3.6.11 below, the Reactor Building purge isolation valves (AH-V-1A&D) shall be limited to less than 31^0 and (AH-V-1B&C) shall be limited to less than 33^0 open, by positive means, while purging is conducted.

3.6.10 During STARTUP, HOT STANDBY and POWER OPERATION:

a. Containment purging shall not be performed for temperature or humidity control.

b. Containment purging is permitted to reduce airborne activity in order to facilitate containment entry for the following reasons:

(1) Non-routine safety-related corrective maintenance.

(2) Non-routine safety-related surveillance.

(3) Performance of Technical Specification required surveillances.

(4) Radiation Surveys.

(5) Engineering support of safety-related modifications for pre-outage planning.

(6) Purging prior to shutdown to prevent delaying of outage commencement (24 hours prior to shutdown).

c. Containment purging is permitted for Reactor Building pressure control.

d. To the extent practicable the above containment entries shall be scheduled to coincide, in order to minimize instances of purging.

3.6.11 When the reactor is in COLD SHUTDOWN or REFUELING SHUTDOWN continuous purging is permitted with the Reactor Building purge isolation valves opened fully.

Bases

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure will build up in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

A condition requiring integrity of containment exists whenever the reactor coolant system is open to the atmosphere and there is insufficient soluble poison in the reactor coolant to maintain the core one percent subcritical in the event all control rods are withdrawn.

The reactor building is designed for an internal pressure of 55 psig, and an external pressure 2.5 psi greater than the internal pressure.

Due to industry reports of elastomer degradation in containment purge valve seats unique action requirements are now designated to help preclude common mode failure of both valves in series. An increased frequency of leak rate testing is also incorporated to help assure timely discovery and resolution of any seat degradation.

An analysis of the impact of purging on ECCS performance and an evaluation of the radiological consequences of a design basis accident while purging have been completed and accepted by the NRC staff. The purge isolation valves have been analyzed capable of closing against the dynamic forces associated with a loss-of-coolant accident when limited to a nominal 30^0 open.

Allowing purge operations during STARTUP, HOT STANDBY and POWER OPERATION (T.S. 3.6.10) is more beneficial than requiring a cooldown to cold shutdown from the standpoint of (a) avoiding unnecessary thermal stress cycles on the reactor coolant system and its components and (b) reducing the potential for causing unnecessary challenges to the reactor trip and safeguards systems.

The recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions 2) radiolytic decomposition of water and 3) corrosion of metals within containment. The recombiner is designed in accordance with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971, the acceptance criteria of the Standard Review Plan (S.R.P.) 6.2.5., and NUREG 0578, July 1979. In addition to the installed hydrogen recombiner, a second recombiner including all piping, electrical, and structural provisions is available on site.

The hydrogen mixing is provided by the reactor building ventilation system to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

Interspace pressurization leak testing of containment purge valves is performed once very three months. The primary objective of this testing per NRC Safety Issue B-24, is to identify excessive degradation of the resilient seats in a timely manner. Upon failing the quarterly test, manual closure of the valve and retesting are performed in order to identify leakage caused by excessive seat degradation. Manual closure means closure of the valve by means other than the normal operator.

REFERENCES

FSAR Section 5.2.2.4.3

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.

*Not required until criticality for Cycle 5 operation.

Bases

The Reactor Building Purge Exhaust System filter AH-F1 is normally used to filter all reactor building exhaust air. It is necessary to demonstrate operability of the 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.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal absorbers to prevent clogging of the iodine adsorbers for all emergency air treatment systems. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accident analyzed in FSAR update Section 14.2.2.1 which assumes 90% efficiency for inorganic iodines and 70% efficiency for organic iodines.

The flow through AH-F1 can vary from 0 CFM to 50,000 CFM, the maximum purge flow rate. During all modes except COLD SHUTDOWN, the purge valves are limited to no more than 30° open (90° being full open). This provides greater assurance of containment isolation dependability per NUREG 0737 Item II.E.4.2 Attachment 1 Item (2)(a). Makeup air is provided between filter AH-F1 and fans AH-E7A and B. (See also T.S. 3.6).

The in-place DOP and halogenated hydrocarbon tests of the filter banks and the laboratory tests of the carbon samples will be done using the test methods and acceptance criteria of Regulatory Guide 1.52 (Rev. 2), except that DOP and Freon tests will be performed such that radiation release limitations are not exceeded.

References

(1) FSAR Section 5.3.3
(2) FSAR Section 5.6
(3) FSAR Section 9.8
(4) Update FSAR Section 14.2.2.1

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
28. Radiation Monitoring Systems*	W(1)(3)	M(3)	Q(2)	(1) Using the installed check source when background is less than twice the expected increase in cpm which would result from the check source alone. Background readings greater than this value are sufficient in themselves to show that the monitor is functioning. (2) Except area gamma radiation monitors RM-G5, RM-G6, RM-G7, and RM-G8 which are located in the Reactor Building. When purging is permitted per T.S. 3.6, RM-G5 will be calibrated quarterly. If purging is not permitted per T.S. 3.6 RM-G5 shall be calibrated at the next scheduled reactor shutdown following the quarter in which calibration would normally be due. RM-G6, RM-G7, and RM-G8 which are in high radiation areas shall be calibrated at the next scheduled reactor shutdown following the quarter in which calibration is due, if a shutdown during the quarter does not occur. (3) Surveillances are required to be performed only when containment integrity is required. This applies to monitors which initiate containment isolation only.
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

* Does not include the monitors covered under specification 3.5.5.2 and 4.1.3

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Specific Activity Determination to compare to the 100/$\overline{E}$μCi/gm limit	At least once each 72 hours during POWER OPERATION, HOT STANDBY, START-UP, and HOT SHUTDOWN.
	b. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	i) 1 per 14 days during power operations. ii) One Sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a one hour period during power operation, start-up and hot standby. iii) # Once per 4 hours, whenever the specific activity exceeds 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/$\overline{E}$μCi/gram during all modes but refueling.
	c. Radiochemical for $\overline{E}$ Determination	1 per 6 months* during power operation
	d. Chemistry (Cl, F and O_2)	5 times/week when T_{avg} is greater than 200°F.
	e. Boron concentration	2 times/week
	f. Tritium Radioactivity	Monthly
2. Borated Water Storage Tank Water Sample	Boron concentration	Weekly and after each makeup when reactor coolant system pressure is greater than 300 psig or T_{avg} is greater than 200°F.
3. Core Flooding Tank Water Sample	Boron concentration	Monthly and after each makeup when RCS pressure is greater than 700 psig.

	Item	Check	Frequency
4.	Spent Fuel Pool Water Sample	Boron concentration	Monthly and after each makeup
5.	Secondary Coolant	a. Gross activity	Weekly when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.
		b. Iodine Analysis**	
6.	Boric Acid Mix Tank or Reclaimed Boric Acid Tank	Boron concentration	Twice weekly***
7,8,9.	Deleted		
10.	Sodium Hydroxide Tank	Concentration	Quarterly and after each makeup.
11.	Deleted		
12.	Condenser Partition Factor	I^{131} Partition Factor	Once if primary/secondary leakage develops, i.e., Gross Beta-Gamma on secondary side of OTSG is greater than 2×10^{-8} microcuries per cc and evidence of fission products is present.

Until the specific activity of the primary coolant system is restored within its limits.

* Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since the reactor was last subscritical for 48 hours or longer.

** When the gross activity increases by a factor of two above background, an iodine analysis will be made and performed thereafter when the gross activity increases by 10 percent.

*** The surveillance of either the Boric Acid Mix Tank or the Reclaimed Boric Acid Tank is not necessary when that respective tank is empty.

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive local leakage, the reactor shall be shutdown and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency.

Local leak detection tests shall be performed at a frequency of at least each refueling period, except that:

a. The equipment hatch and fuel transfer tube seals shall be tested every other refueling period but in no case at intervals greater than 3 years. If they are opened they will be tested after being closed.

b. The entire personnel and emergency airlocks shall be tested once every six months. When the airlocks are opened during the interim between six month tests, the airlock door resilient seals shall be tested within 72 hours of the first of each of a series of openings. This requirement exists whenever containment integrity is required.

c. The reactor building purge isolation valves shall be leak tested each refueling interval per 10 CFR 50 Appendix J, Item III.D.2.

d. An interspace pressurization test (See T.S. 4.4.1.7.1) shall be performed for reactor building purge isolation valves every 3 months. This requirement is not in effect during cold shutdown.

e. Readings of the rotameters in each manifold of the penetration pressurization system shall be recorded at periodic intervals not to exceed three months.

4.4.1.3 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The valves not stroked every three months shall be stroked during each refueling period.

4.4.1.4 Annual Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed annually and prior to any integrated leak test to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical, prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.5 and 4.4.1.2.3, respectively.

4.4.1.6 Operability of Access Hatch Interlocks

1. At least once per 6 months the operability of the personnel and emergency hatch door interlocks and the associated control room annunciator circuits shall be determined. If the interlock permits both doors to be open at the same time or does not provide accurate status indication in the control room, the interlock shall be declared inoperable.

2. During periods when containment integrity is required and an interlock is inoperable, each entry and exit via that airlock shall be locally supervised by a member of the unit-operating maintenance or technical staffs, to assure that only one door is open at any time and that both doors are properly closed following use. A record of supervision and verification of closure shall be maintained during periods of interlock inoperability in an appropriate station log.

3. If an interlock is inoperable for more than 14 days following determination of inoperability, use of the airlock, except for emergency purposes, shall be suspended until the interlock is returned to operable status.

4.4.1.7 Operability of Purge Valves

1. A periodic pressurization of the purge valve interspaces to 50.6 psig per T.S. 4.4.1.2.5d shall be performed to help assure timely detection and resolution of valve and/or actuator degradation. The acceptance criteria is that total local leakage when updated for the new purge valve leakage shall be less than $0.6L_A$. See Tech Spec 3.6.8 for further action.

2. The rubber seats on purge valves shall be visually examined each refueling interval to detect degradation (e.g. cracking, brittleness, etc.) and to assure timely cleaning, lubrication, and seat replacement. As a minimum seats shall be replaced at the first refueling following 5 years of seat service.

Bases(1)

The reactor building is designed for an internal pressure of 55 psig and a steam-air mixture temperature of 281F. Prior to initial operation, the containment was strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment was also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests established the acceptance criteria of 4.4.1.1.3.

The performance of periodic integrated and local leakage rate tests during the plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In

order to provide a realistic appraisal of the integrity of the containment under accident conditions "as found" local leakage results must be documented for correction of the integrated leakage rate test results. Containment isolation valves are to be closed in the normal manner prior to local or integrated leakage rate tests.

The minimum test pressure of 27.5 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at the reduced pressure. The specification provides a relationship for relating the measured leakage of air at the reduced pressure to the potential leakage of 55 psig. The minimum of 24 hours was specified for the integrated leakage rate test to help stabilize conditions and thus improve accuracy and to better evaluate data scatter. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.10 percent leakage rate at 55 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation, and the low value (0.06 percent) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the reactor building liner due to the mechanical closure involved. The basis for specifying a total leakage rate of 0.06 percent from those penetrations and isolation valves is that more than one-half of the allowable integrated leakage rate will be from these sources.

Valve operability tests are specified to assure proper closure or opening of the reactor building isolation valves to provide for isolation or functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

Periodic surveillance of the airlock interlock system is specified to assure continued operability and preclude instances where one or both doors are inadvertently left open. When an airlock is inoperable and containment integrity is required, local supervision of airlock operation is specified.

Purge valve interspace pressurization test operability requirements and inspections provide a high degree of assurance of purge valve performance as containment isolation barriers.

References

(1) FSAR, Section 5

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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 2 years, 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.

*Surveillance to be performed prior to Cycle 5 criticality in lieu of once per refueling interval or once per 2 years.

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.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with approved test procedures. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable all adsorbent in the system should be replaced with an adsorbent qualified according to Regulatory Guide 1.52, March 1978. Tests of the HEPA filters with DOP aerosol shall also be performed in accordance with approved test procedures. Any HEPA filters found defective should be replaced with filters qualified according to Regulatory Guide 1.52, March 1978.

Fans AH-E7A&B performance verification is necessary to ensure adequate flow to perform the filter surveillance of T.S. 4.12.2.1 and 4.12.2.3 and can only be demonstrated by running both fans simultaneously. This can only be accomplished when purge valves are not limited to 30^0 open (i.e., cold shutdown).

Since H_2 purge has been superseded by the installation of H_2 recombiners at TMI-1, the reactor building purge exhaust system no longer is relied upon to serve an operating accident mitigating (i.e. LOCA) function. The retest requirement of T.S. 4.12.2.2a has therefore been changed to reflect the same retest requirements as the auxiliary and fuel handling building ventilation system which similarly serves no operating accident mitigating function.

If significant painting, steam, fire, or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the Operations and Maintenance Director - TMI-1.

4.18.6 HOSE STATIONS

Applicability: Hose stations listed in Table 3.18-2.

Objective: To insure system operability.

Specification:

4.18.6.1 Each fire hose station shall be verified operable:

a. At least once per month* by visual inspection of the station to assure all equipment is at the station.

b. At least once per 18 months* by removing the hose for inspection and re-racking, and replacing all gaskets in the couplings that are degraded.

c. At least once per 3 years, partially open hose station valves to verify valve operability and no blockage.

d. At least once per 3 years by conducting a hose hydrostatic test at a pressure at least 50 psi greater than the maximum pressure available at that hose station.

* For hose stations in the Reactor Building these inspections may be deferred, if purging is not permitted per TS 3.6, until the first shutdown greater than 48 hours following the interval which permits purging.

(2)	Steam Generator Tube Inspection Program (See section 4.19.5)	within 3 months after completion of inspection.
(3)	Containment Integrated Leak Rate Test	within 6 months after completion of test.
(4)	Inservice Inspection Program	within 6 months after five years of operation.
(5)	Radioactive Sealed Source Leakage Test revealing the presence of ≥ 0.005 microcuries of Removable Contamination.	within 90 days after completion of test.
(6)	Special Report - Exceeding 500 hrs. of operation with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 in any 6 month period. Indicate number of hours of operation above this limit. See T.S. 3.1.4	submit within 30 days.

6.9.4 ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING 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 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.

6.9.4.2 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. If harmful effects of evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of the Radiological Assessment BTP on the REMP March 1978 of all radiological environmental samples taken during the report period. 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 of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 108 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/Evaluation

In order to complete the resolution of mutliplant action (MPA) B-24 regarding purge and vent, the licensee was requested to provide certain Technical Specification (TS) changes. Guidance was provided in Standard Technical Specifications (STSs) sent to the licensee in letters from J. Stolz (NRC) to H. Hukill (GPUN) dated April 30, 1982, and July 8, 1983. The licensee has responded with Technical Specification Change Request (TSCR) 116 dated August 13, 1982, TSCR 116, Rev. 1, dated November 24, 1983, and supplements to TSCR 116, Rev. 1 dated June 5, and December 3, 1984. We have reviewed the licensee's submittals for compatibility with the resolution of MPA B-24 items using the STSs as review guidance. The two subjects covered by the STSs are (1) primary coolant activity and (2) vent/purge valve operability and surveillance.

The licensee found that other changes were needed in the TMI-1 TSs for compatibility between the proposed TS requirements and operational needs. These changes were not covered by STSs and therefore were reviewed on a plant-specific basis (Section 3, following).

1. Primary coolant activity (proposed TS pages 3-8, 3-9, 3-9a, 3-9b, 4-9 and 6-17).

Background

The NRC staff's generic safety evaluation of the radiological consequences of purging and venting at power concluded that 10 CFR Part 100 dose guidelines and the Standard Review Plan requirements would be met if iodine equilibrium and spiking coolant activity levels are limited to those of the STSs.

Evaluation

The licensee has proposed revisions to TS 3.1.4, Reactor Coolant Activity, and TS 4.1, Operational Safety Review (Table 4.1-3, Minimum Sampling Frequency for Reactor Coolant). The proposed TSs would limit the allowable specific activity of the primary coolant to not more than

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100/E microcuries/gm and generally to not more than 1.0 microcuries/gm DOSE EQUIVALENT. These proposed limits, and the proposed action to be followed if the limits are exceeded, including reporting requirements, are consistent with the STSs and therefore are acceptable. That is, the revisions proposed for TS pages 3-8, 3-9, 3-9a and 3-9b are acceptable.

The proposed sampling program for reactor coolant (TS Table 4.1-3) would require checking of specific activity for comparison with the 100/E microcurie/gm limit each 72 hours (except during cold shutdown) and, in general, an isotopic analysis for DOSE EQUIVALENT I-131 concentration every 14 days. These requirements along with the other limitations that would be imposed by the TSs are compatible with the STSs and therefore acceptable. That is, proposed revisions to TS page 4-9 are acceptable.

In proposed TS 6.9.3, paragraph (6) has been added to require a special report to the NRC if the DOSE EQUIVALENT I-131 exceeds 1.0 microcuries/gm for more than 500 hours in a 6-month period. This requirement on TS page 6-17 is consistent with the STSs and therefore acceptable.

2. Vent/purge valve operability and surveillance requirements (TS pages 3-41, 3-41a, 3-41b, 3-41c, 4-34, 4-34a and 4-34b).

Background

The satisfactory resolution of MPA B-24 required operability and surveillance requirements for purge/vent valves that would assure reliability and the ability to close within the required time.

Evaluation

Proposed TS 3.6.6 addresses reactor building isolation valves other than (large) purge valves and would require one valve to be closed if the companion valve in-line is inoperable, or proceed to cold shutdown.

Inoperability of the large purge valves because of resilient seal leakage is covered by proposed TS 3.6.8. The proposed actions would require the reactor to be shut down if excessive leakage cannot be corrected within 72 hours following quarterly testing. The proposed TS actions are consistent with the STSs and resolve the seal leakage concerns identified in MPA B-24.

The limits on purge valve opening that were found acceptable in the resolution of MPA B-24 have been included in proposed TS 3.6.9. Therefore this section is acceptable.

It is our position that purging should be limited during reactor operation because a closed containment is inherently safer than an open containment during purging. Some licensees have a specific limit on purging but for TMI-1 a more flexible TS is proposed that identifies acceptable reasons for purging (proposed TS 3.6.10). We find that the listed reasons and the licensee's commitment to minimize entries (and thus purging) are acceptable. The licensee has estimated that about 100 hours of purging per month will be needed and has committed to

provide a report after the next cycle of operation that discusses the actual amount of purging during operation. During cold shutdown, continuous purging is permitted and therefore proposed TS 3.6.11 is acceptable and in accordance with the resolution of MPA B-24.

The proposed TSs on purge valve testing frequency (TS 4.4.1.2.5) would require testing each 3 months, and the proposed TSs on operability criteria (TS 4.4.1.7) specify acceptance criteria and rubber seat (resilient seal) inspection requirements, as well as periodic seat replacement requirements. These requirements meet the guidelines of MPA B-24. In particular, the acceptance criteria provide reasonable assurance that gross seal deterioration will be detected in a timely manner, and the seal replacement frequency (at least every 5 years) is in accordance with the manufacturer's recommendations. Accordingly, we find these proposed TSs acceptable.

The licensee has not specified valve closure time as an operability requirement in the TSs because closure times are included in the inservice testing program. We find this acceptable because the licensee has committed to include a discussion of purge valve closure time in the Final Safety Analysis Report (FSAR) including the safety-related reasons for needing less than 3.5 sec. closure time, i.e., to mitigate damage to downstream structures/components following a Loss-of-Coolant Accident (LOCA).

3. Other Changes Initiated by Licensee

3.1 - Reactor building purge air treatment system (TS pages 3-62a, 3-62b, 4-55b and 4-55c)

Background

The licensee has installed a hydrogen recombiner and has completed provisions for a second hydrogen recombiner. Therefore purging is no longer needed for hydrogen control and the reactor building purge air treatment system does not perform an operating accident mitigation function; it is required only for purging associated with normal releases and for fuel handling accidents in containment. Therefore, the system fans are no longer required to be operable during normal plant operation but they must operate within design limits to properly test the filters. The filters, however, are needed to mitigate accident releases whenever the purge valves are open during refueling operations.

Evaluation

Proposed TSs 3.15.2.1 and 3.15.2.3 remove operability requirements for fans AH-E7A and B. Additionally, surveillance requirements for fans (TS 3.14.2.2(e)) are moved to TS 4.12.2.2.e. These changes are acceptable because the fans are no longer required during normal operations since the hydrogen recombiner will be available as discussed under background, above. The proposed surveillance requirements for fans (TS 4.12.2.2.e) would require testing of fans each refueling (or 2 years) in lieu of 10 hours of operations each month. This change is acceptable because the

fans will be required only for filter testing specified in TS 4.12.2.2 which is done primarily at refueling.

Proposed TS 3.15.2.1 also clarifies that filter operability is not required when the purge valves are closed. This change is acceptable because when purge valves are closed, no filtering is needed.

In accordance with proposed TS 3.15.2.3, purging will not now be permitted with the filter inoperable and the purge valves open (previously permitted for 30 days). This change is acceptable because it assures compliance with 10 CFR Part 100 release guidelines during refueling operations.

The bases for proposed TS 3.15.2 would be revised to reflect the changed requirements for the reactor building purge air treatment system because of the availability of hydrogen recombiners. Also, the bases for filter testing would be revised to eliminate considerations of purge valve position because testing will be done at cold shutdown when the purge valves are fully open and the fans can operate at full flow.

In proposed TS 4.12.2 the surveillance interval for the reactor building purge air treatment system would be extended from a maximum of 18 months to a maximum of 2 years. This change would make the requirements for the reactor building compatible with those for auxiliary and fuel handling buildings whose air treatment systems serve no operating accident mitigation function. All these systems will now serve similar functions because of the availability of the hydrogen recombiners for the reactor building ventilation system, and therefore the proposed change is acceptable.

In proposed TS 4.12.2.2.a the reactor building air treatment system surveillance would be required within 30 days prior to movement of fuel in lieu of surveillance each 720 hours of operation. The 720-hour surveillance requirement is no longer needed for this air treatment system since hydrogen recombiners are available to control hydrogen in the containment.

The proposed TS 4.12.2.3 would require air distribution tests on the HEPA filter bank to be performed at the full reactor building purge system flow rate of 50,000 cfm rather than 25,000 cfm as currently provided. This section also would clarify that testing would be performed with the purge valves fully open. The proposed change is appropriate and acceptable because the system should be tested at maximum air flow to assure operability under all anticipated conditions.

3.2 - Calibration of containment area monitors (TS page 4-5a)

Background

An area radiation monitoring system is designed to (1) inform operations personnel of radiation levels in areas where these systems are located, (2) provide warning (by audible and visual alarms) both locally and in the control room, when abnormal levels occur, (3) warn of possible equipment malfunction, and (4) provide a continuous record of radiation

levels at key locations. The criteria for area monitor location are usually based on occupancy factors and potential exposure to high radiation, among other factors. ANSI 6.8.1-1981 "Location and Design Criteria for Area Radiation Monitoring Systems for Light Water Nuclear Reactors" specifies that area monitors should be used to assist in minimizing exposure to personnel and shall be located in areas most likely to be occupied.

Based on the criteria as stated above, containment area radiation monitors function essentially as event oriented monitors to provide functions (2), (3) and (4) rather than personnel monitors used to assist in minimizing exposure to personnel. This is because no personnel entry can be made into containment, as with all high radiation areas, without the administrative control TS requirement that specifies Radiation Work Permits (RWPs) for entry into a high radiation area. Additionally, the TS requires that portable survey meters or other monitoring devices, which continuously indicate or integrate the radiation dose rate in the area of concern, must be used. Thus, the area monitors, if they are in the appropriate location where work is to be performed, merely serve as general level indicators essentially as a secondary line of defense to the portable survey meters.

It should be noted that calibration of containment monitors, such as area gamma radiation monitor RM-G5, may be considered a special case when compared to the calibration needs of all other area monitors in the plant. Calibration of these monitors requires personnel entry to the containment. With the reactor at power, this person would be potentially exposed to high levels of neutron and gamma radiation, as well as airborne radioactivity levels. In order to minimize radiation to personnel calibrating the containment area monitors, the licensee would have to shut down the reactor and purge containment prior to personnel entry. However, shutting down the reactor every quarter to calibrate the containment area monitors is not cost effective with respect to the philosophy of ALARA.

Some additional factors that would influence our evaluation are as follows: Area monitors in containment are not safety related and are not required to operate during accident or seismic events. Only those two high-range monitors installed in accordance with NUREG-0737 need be so qualified. Three Mile Island Unit 1 TSs require all area monitors to have a channel check and functional test at frequent intervals (i.e., weekly and monthly respectively) to verify operability of the channel, including alarm function.

Evaluation

From the above, we can draw the following conclusions concerning area monitor RM-G5, and area monitors in containment in general:

(1) They are not used primarily for occupational exposure purposes.

(2) They are located in high radiation areas so that an RWP and portable radiation survey equipment are required for entry into the area in accordance with TSs. Therefore, these area monitors are not needed for radiation protection when entry into containment is made.

(3) There is no need for calibrating them every quarter if this necessitates unnecessarily exposing personnel to high levels of radiation during each calibration. This is not consistent with ALARA philosophy. Channel checks and functional tests are performed frequently to assure operability of the channel in accordance with TSs.

(4) They should, however, be calibrated quarterly whenever full containment purging is permitted and not less often than once every 18 months during refueling outages.

(5) Recalibration of these monitors should be performed after maintenance or replacement of any component that could affect calibration.

Based on the above criteria, we concur with the TS change that defers calibration frequency requirements for area monitor RM-G5 from quarterly to periods of shutdown when full purging is permitted.

3.3 - Removal of hydrogen purge system (TS pages (iii), 4-37 and 4-38)

Background

TS 4.4.3 describes the surveillance procedures and limiting conditions of operation that were associated with the safety-related post-accident hydrogen purge system.

The licensee proposes to delete Section 4.4.3 from the TSs because the licensee has installed a hydrogen recombiner and has provisions for a second recombiner to remove the hydrogen that could be generated following a LOCA.

Evaluation

Because the licensee is replacing the hydrogen purge system with hydrogen recombiners, the hydrogen purge system is no longer needed to perform a safety-related function. Therefore, we agree that it is appropriate and acceptable to remove the testing and availability requirements of the hydrogen purge system from the TSs.

3.4 - Surveillance of fire hose stations (TS page 4-76)

Background

The existing TSs require monthly visual inspections of fire hose stations and, each 18 months, removal and inspection of hoses and gaskets. The licensee proposes to permit exceptions to delay these surveillances if the reactor building is not purged and therefore the hose stations are not accessible.

Evaluation

We find the proposed exception acceptable because of the likelihood that the required surveillances would not be delayed and that any delay would be short, considering that the licensee anticipates about 100 hours per month of purging. Therefore, the proposed changes to TS 4.18.6.1 are acceptable.

3.6 - Administrative changes (TS page 4-55c)

An administrative change is proposed on TS page 4-55c where the title "Unit Superintendent" is changed to reflect the revision of Amendment 77 to read "Operations and Maintenance Director - TMI-1." This change is acceptable because it merely updates the TSs.

Environmental Considerations

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The amendment also relates to a change in reporting requirements. 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 offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission 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, this 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 the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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