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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/$\overline{E}$ 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 months 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/$\overline{E}$ microcuries/gram be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when primary coolant activity is less than 100/$\overline{E}$ 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/$\overline{E}$ microcuires/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. This report shall contain the results of the specific activity analyses together with the following information:

* $\overline{E}$ 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 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.

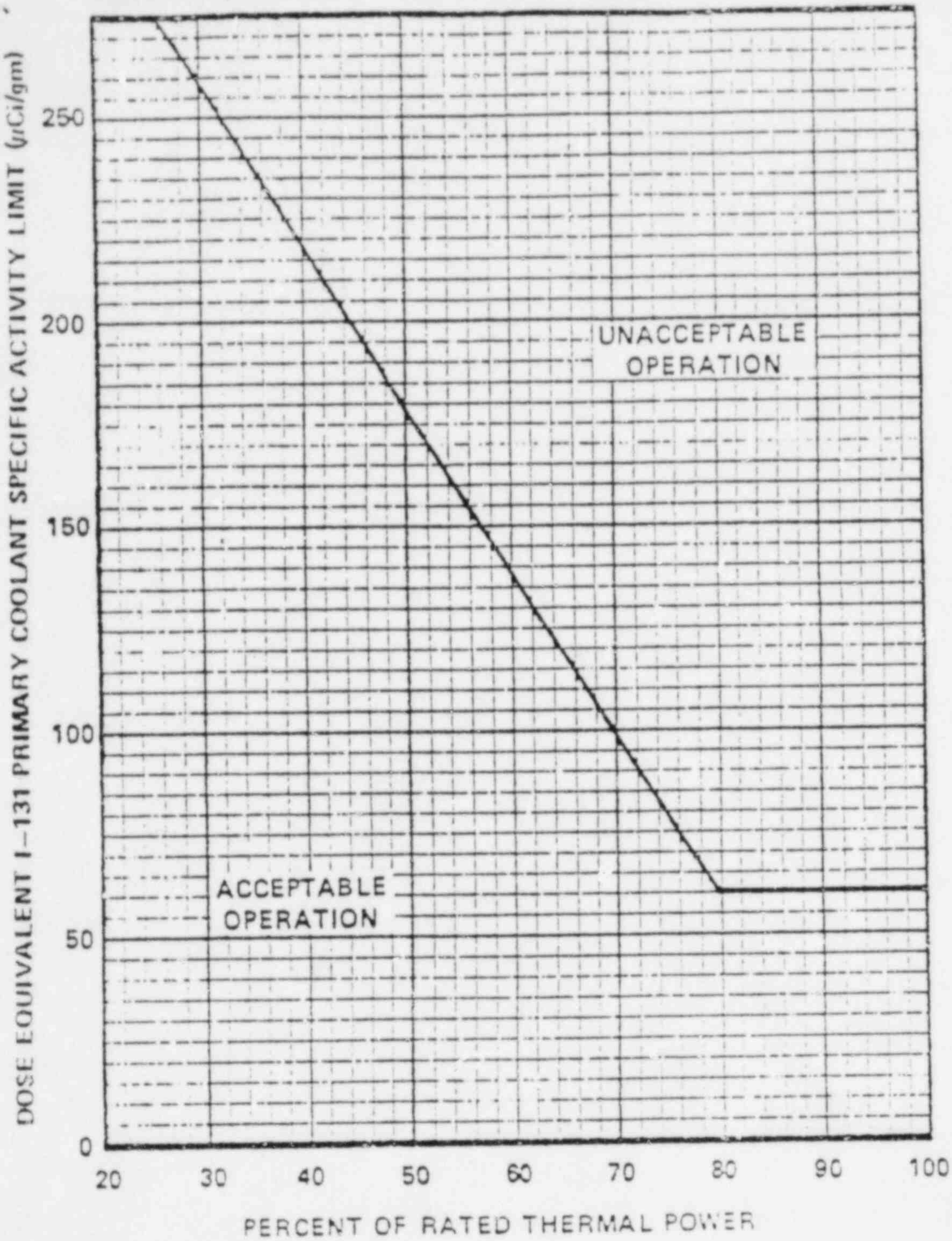

FIGURE 3.1-2a

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.

3.6 REACTOR BUILDING

Applicability

Applies to the containment integrity of the reactor building.

Objective

To assure containment integrity during startup and operation.

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 that 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 While containment integrity is required (See TS 3.6.1), if a 48" reactor building purge valve is found to be INOPERABLE (See TS 4.4.1.7) immediately close the associated valve and within 24 hours verify that the associated valve is OPERABLE per T.S. 3.6.8.

3.6.8 If INOPERABILITY was due to excessive combined leakage (See TS 4.4.1.7.1) demonstrate that one of the valves has no detectable leakage (by using soap solution at equal to or greater than 5 psig interspace pressure) and that this valve is maintained closed by approved administrative controls. This shall be sufficient verification that the one valve is OPERABLE.

Plant operation may then continue provided that the OPERABLE valve is verified to be closed at least once per 31 days and is maintained closed by approved administrative controls.

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.

3.6.9 Except as specified in 3.6.10 below the Reactor Building purge isolation valves shall be limited to less than 31.75° (AH-V-1A & D) and less than 33.29° (AH-V-1B & C) open, by positive means, while purging is conducted. If all technical specification requirements applicable to the purge system have been satisfied, purging of the containment is permitted.

3.6.10 When the reactor is in cold shutdown the Reactor Building purge isolation valves may be 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 reports of unsatisfactory performance of resilient seats of containment purge isolation valves throughout the nuclear industry, a leakage test program has been implemented for these valves. This program assures a higher degree of assurance of purge valve operability.

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 demonstrated capable of closing against the dynamic forces associated with a loss-of-coolant accident when limited to 30° open.

Allowing purge operations during hot shutdown and operation (T.S. 3.6.9) 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.

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 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	

TABLE 4.1-3

MINIMUM SAMPLING FREQUENCY

Item	Check	Frequency
1. Reactor Coolant	a. Specific Activity Determination to compare to the $100/\overline{E}\mu Ci/gm$ limit	At least once each 72 hours during power operation, start-up, hot standby, 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 EQIVALENT 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
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 Tavg 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.

4.	Spent Fuel Pool Water Sample	Boron concentration	Monthly and after each makeup
5.	Secondary Coolant	a. 15 min. gross degassed Beta-Gamma Activity b. Iodine Analysis **	Weekly when reactor coolant system pressure is greater than 300 psig or Tav is greater than 200°F.
6.	Boric Acid Mix Tank or Reclaimed Boric Acid Tank	Boron concentration	Twice weekly
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 subcritical 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.

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 when purging is permitted (TS 3.6). This requirement 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 refueling or once per 6 months if reactor building purging is permitted per TS 3.6, 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 (penetrations and isolation valves which are not continuously pressurized by the penetration pressurization system or are not fluid blocked post-accident by the fluid block system) 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. Particular attention is given to testing those penetrations and process lines not serviced by the penetration pressurization system or the fluid block system. 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 ($\pm$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 $\pm$ 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 $\pm$20%. The test shall be performed at 50,000 cfm ($\pm$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 can only be demonstrated by running both fans simultaneously. This can only be accomplished when purge valves are not limited to 30° 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.

4.17 Shock Suppressors (Snubbers)

Applicability

Applies to the inspection of hydraulic snubbers listed in Table 3.17.1 to determine their operability.

Objective

To provide assurance of the operability of the hydraulic snubbers.

Specification

4.17.1 All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing, or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include but not necessarily be limited to, inspection of hydraulic fluid reservoir, fluid connections, and linkage connections to the piping and anchor to verify snubber operability in accordance with the following schedule except those in the Reactor Building when purging is not permitted per T.S. 3.6,* shall be inspected during each shutdown greater than 48 hours which permits purging following the schedule specified below.

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
$\geq$ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

These two groups may be inspected independently according to the above schedule.

4.17.2 All accessible hydraulic snubbers whose seal materials are other than ethylene propylene or other material that has been demonstrated to be compatible with the operating environment shall be visually inspected for operability at least every 31 days. When purging is not permitted per TS 3.6, snubbers in the Reactor Building are considered inaccessible and shall be inspected during each shutdown greater than 48 hours which permits reactor building purging in lieu of each 31 days unless previously inspected within 31 days of the shutdown.

4.17.3 For the purpose of entering the schedule in Specification 4.17.1, the initial inspection interval shall be 12 months ±25%.

4.17.4 Once each refueling cycle, a representative sample of ten hydraulic snubbers or approximately 10% of the hydraulic snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement, lockup and bleed. For each unit and subsequent unit found inoperable, an additional 10% or ten hydraulic snubbers, whichever is less, shall be so tested until no more failures are found or all units have been tested. Snubbers of rated capacity greater than 50,000 lbs. need not be functionally tested.

Bases

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection will include verification of proper orientation, adequate hydraulic fluid level, and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at operating facilities has shown that the required surveillance program should assure an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. However, based upon the results of snubber inspection at TMI-1 and engineering analyses, GPU Nuclear may propose for NRC review and approval, an alternative program for snubber inspection which will provide assurance of an equivalent level of snubber performance.

The initial inspection interval for visual inspection is based upon the results of the inspection performed during the March-April, 1977 refueling outage.

Snubbers containing seal material which has not been demonstrated by operating experience, lab tests, or analysis to be compatible with the operating environment should be inspected more frequently (every month) until material compatibility is confirmed or an appropriate changeout is completed. Due to human health considerations (i.e., ALARA), snubbers in the Reactor Building are inspected only during shutdown when Reactor Building purging is not permitted per T.S. 3.6.

Examination of defective snubbers at reactor facilities and material tests performed at several laboratories (Reference 1) has shown that millable gum

polyurethane deteriorates rapidly under the temperature and moisture conditions present in many snubber locations. Although molded polyurethane exhibits greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. Lab tests and in-plant experience indicate that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installation.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers designated in Table 3.17.1 as being in high radiation areas, or those especially difficult to remove, need not be selected for functional tests provided operability was previously verified.

Snubbers of rated capacity greater than 50,000 lbs. are exempt from the functional testing requirements because of the impracticality of testing such large units.

Reference

(1) Report H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974

Subject: Hydraulic Shock Sway Arrestors

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 defered, 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 $\geq$ 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