TABLE OF CONTENTS

TABLE OF CONTENTS

Amendment No. ~~11, 22, 30, 41, 47, 55, 72, 78, 95, 97, 119, 122, 129,~~
~~137, 146, 197, 242, 245~~, 246

1.6 POWER DISTRIBUTION

1.6.1 QUADRANT POWER TILT

Quadrant power tilt is defined by the following equation and is expressed in percent.

$$100\left[\frac{\textit{Power in Any Core Quadrant}}{\textit{Average Power of All Quadrants}}-1\right]$$

The quadrant tilt limits are stated in Specification 3.5.2.4.

1.6.2 AXIAL POWER IMBALANCE

Axial power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

1.7 CONTAINMENT INTEGRITY

CONTAINMENT INTEGRITY exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel and emergency air locks are closed and sealed.

b. All passive Containment Isolation Valves (CIVs) and isolation devices, including manual valves and blind flanges, are closed as required by the "Containment Integrity Check List" attached to the operating procedure, "Containment Integrity and Access Limits." Normally closed passive CIVs may be unisolated intermittently under administrative control.

c. All active CIVs, including power-operated valves, check valves, and relief valves, are OPERABLE or locked closed. Normally closed active CIVs (other than the purge valves) may be unisolated intermittently or manual control of power-operated valves may be substituted for automatic control under administrative control.

d. The containment leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 FIRE SUPPRESSION WATER SYSTEM

A FIRE SUPPRESSION WATER SYSTEM shall consist of: a water source, gravity tank or pump and distribution piping with associated sectionalizing control or isolation valves. Such valves include yard hydrant curb valves, and the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

3.1.6 LEAKAGE

Applicability

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

Objective

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

Specification

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

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

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

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

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

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

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

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

Bases (Continued)

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

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

Assuming the existence of the maximum allowable activity in the reactor coolant, a reactor coolant leakage rate of less than one gpm unidentified leakage within the reactor or auxiliary building or any of the closed cooling water systems indicated above, is a conservative limit on what is allowable before the dose rate limits of the ODCM would be exceeded.

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

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

Bases (Continued)

The unidentified leakage limit of 1 gpm is established as a quantity which can be accurately measured while sufficiently low to ensure early detection of leakage. Leakage of this magnitude can be reasonably detected within a matter of hours, thus providing confidence that cracks associated with such leakage will not develop into a critical size before mitigating actions can be taken.

Total reactor coolant leakage is limited by this specification to 10 gpm. This limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of unidentified leakage.

The primary to secondary leakage through the steam generator tubes is limited to 1 gpm total. This limit ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of a steam line break. Steam generator leakage is quantified by analysis of secondary plant activity.

If reactor coolant leakage is to the auxiliary building, it may be identified by one or more of the following methods:

a. The auxiliary and fuel handling building vent radioactive gas monitor is sensitive to very low activity levels and would show an increase in activity level shortly after a reactor coolant leak developed within the auxiliary building.

b. Water inventories around the auxiliary building sump.

c. Periodic equipment inspections.

d. In the event of gross leakage, in excess of 13 gpm, the individual cubicle leak detectors in the makeup and decay heat pump cubicles, will alarm in the control room to backup "a", "b", and "c" above.

When the source and location of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by TMI-1 Plant Operations.

3.5.2 CONTROL ROD GROUP AND POWER DISTRIBUTION LIMITS

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall not be less than one percent ΔK/K with the highest worth control rod fully withdrawn.

3.5.2.2 Operation with inoperable rods:

a. Operation with more than one inoperable rod as defined in Specification 4.7.1 in the safety or regulating rod banks shall not be permitted. Verify SDM $\geq$ 1% Δk/k or initiate boration to restore within limits within 1 hour. The reactor shall be brought to HOT SHUTDOWN within 6 hours.

b. If a control rod in the regulating and/or safety rod banks is declared inoperable in the withdrawn position as defined in Specification Paragraph 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of one percent Δk/k hot shutdown margin. Boration may be initiated to increase the available rod worth either to compensate for the worth of the inoperable rod or until the regulating banks are fully withdrawn, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

c. If within one hour of determination of an inoperable rod as defined in Specification 4.7.1, and once per 12 hours thereafter, it is not determined that a one percent Δk/k hot shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the HOT SHUTDOWN condition within 6 hours until this margin is established.

d. Following the determination of an inoperable rod as defined in Specification 4.7.1, all rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

e. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, and cannot be aligned per 3.5.2.2.f, power shall be reduced to $\leq$ 60% of the thermal power allowable for the reactor coolant pump combination within 2 hours, and the overpower trip setpoint shall be reduced to $\leq$ 70% of the thermal power allowable within 10 hours. Verify the potential ejected rod worth (ERW) is within the assumptions of the ERW analysis and verify peaking factor ($F_Q(Z)$ and $F_{\Delta H}^{N}$) limits per the COLR have not been exceeded within 72 hours.

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 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, CONTAINMENT INTEGRITY (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 degrees F or greater.

c. Nuclear fuel is in the core.

3.6.2 Except as provided in Specifications 3.6.6, 3.6.8, and 3.6.12, 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 (CIVs) which should be closed are closed and are conspicuously marked.

3.6.6 When CONTAINMENT INTEGRITY is required, if a CIV (other than a purge valve) is determined to be inoperable:

a. For lines isolable by two or more CIVs, the CIV(s)* required to isolate the penetration shall be verified to be OPERABLE. If the inoperable valve is not restored within 48 hours, at least one CIV* in the line 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.

b. For lines isolable by one CIV, where the other barrier is a closed system, the line shall be isolated by at least one closed and de-activated automatic valve, closed manual valve, or blind flange within 72 hours 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.

* All CIVs required to isolate the penetration.

3.6 REACTOR BUILDING (Continued)

3.6.7 DELETED

3.6.8 While CONTAINMENT INTEGRITY is required (see Specification 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 Specification 6.8.5), 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 Specification 6.8.5 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 in accordance with the Reactor Building Leakage Rate Testing Program. 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 degrees and (AH-V-1B&C) shall be limited to less than 33 degrees 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.

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.

The primary Containment Isolation Valves (CIVs) are identified in UFSAR Table 5.3-2. Additional vent, drain, test and other manually operated valves which complete the containment boundary are identified in the containment integrity checklist. For the purpose of this specification, check valves and relief valves identified in the containment integrity checklist are defined to be active valves.

The loss of redundant capability for containment isolation is limited for all penetrations after which the containment penetration must be isolated. Isolation of certain penetrations may require the closure of multiple CIVs due to piping branches.

1. When one of two CIVs in a line is inoperable, the capability to isolate the penetration using the other CIV in the line is promptly verified and at least one valve in the line must be closed within 48 hours or the plant must commence shut down.

2. For those CIVs where the second barrier is a closed system within the Reactor Building, there is no other CIV to isolate the penetration. If operability cannot be regained, the valve must be closed within 72 hours or the plant must commence shut down. An action time of 72 hours is reasonable considering the relative stability of the closed system (hence, reliability) to act as a containment isolation boundary and the relative importance of supporting containment integrity.

The definition of Containment Integrity permits normally closed CIVs, except for the 48 inch purge valves, to be unisolated intermittently or manual control to be substituted for automatic control under administrative control. Administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment (Reference 1). The dedicated individual can be responsible for closing more than one valve provided that the valves are in close vicinity and can be closed in a timely manner. Due to the size of the containment purge line penetration and the fact that those penetrations exhaust directly from the containment atmosphere to the environment, the containment penetrations containing these valves may not be opened under administrative control.

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. Analysis has demonstrated that a purge isolation valve is capable

BASES (Continued)

of closing against the dynamic forces associated with a LOCA when the valve is limited to a nominal 30 degree open position.

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 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.

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J (Reference 2), and the Reactor Building Leakage Rate Testing Program. Each air lock door has been designed and is tested to certify its ability to withstand a pressure in excess of the maximum expected pressure following a Design Basis Accident (DBA) in containment. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events.

Entry and exit is allowed to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed to repair. If the inner door is the one that is inoperable, however, then a short time exists when the containment boundary is not intact (during access through outer door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit the OPERABLE door must be immediately closed. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock. With both air locks inoperable due to inoperability of one door in each of the two air locks, entry and exit is allowed for use of the air locks for 7 days under administrative controls. Containment entry may be required to perform Technical Specifications (TS) Surveillance and Required Actions, as well as other activities on equipment inside containment that are required by TS or activities on equipment that support TS-required equipment. This is not intended to preclude performing other activities (i.e., non-TS-required activities) if the containment was entered, using the inoperable air lock, to perform an allowed activity listed above. This allowance is acceptable due to the low probability of an event that could pressurize the containment during the short time that the OPERABLE door is expected to be open.

With one or more air locks inoperable for reasons other than those described in 3.6.12 "b" or "c," Section 3.6.12.d requires action to be immediately initiated to evaluate previous combined leakage rates using current air lock test results. An evaluation is acceptable since it is overly conservative to immediately declare the containment inoperable if both doors in an air lock have failed a seal test or the overall air lock leakage is not within limits. In many instances (e.g., only one seal per door has failed), containment remains OPERABLE, yet only 1 hour would otherwise be provided to restore the air lock to OPERABLE status prior to requiring a plant shutdown. In addition, even with both doors failing the seal test, the overall containment leakage rate can still be within limits.

BASES (Continued)

Section 3.6.12.d requires that one door in the affected containment air locks(s) must be verified to be closed within 1 hour. Additionally, the affected air lock(s) must be restored to OPERABLE status within the 24 hour Completion Time. 24 hours is considered reasonable for restoring an inoperable air lock to OPERABLE status assuming that at least one door is maintained closed in each affected air lock.

References

(1) NRC Generic Letter 91-08
(2) 10 CFR 50, Appendix J.

TABLE 4.1-2

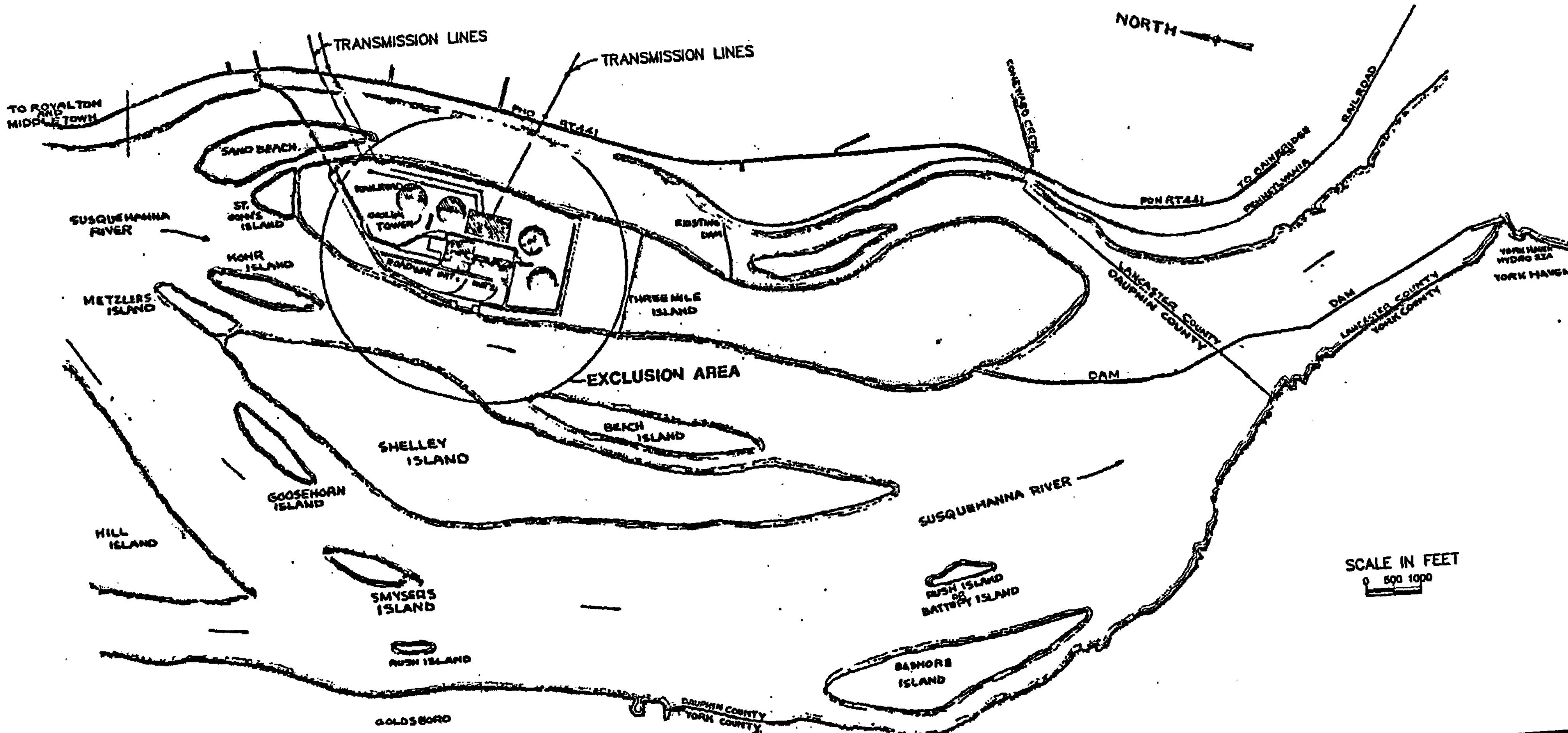
MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rods	Rod drop times of all full length rods	Each Refueling shutdown
2. Control Rod Movement	Movement of each rod	Every 92 days, when reactor is critical
3. Pressurizer Safety Valves	Setpoint	In accordance with the Inservice Testing Program
4. Main Steam Safety Valves	Setpoint	In accordance with the Inservice Testing Program
5. Refueling System Interlocks	Functional	Start of each refueling period
6. (Deleted)	--	--
7. Reactor Coolant System Leakage	Evaluate	Daily, when reactor coolant system temperature is greater than 525 degrees F
8. (Deleted)	--	--
9. Spent Fuel Cooling System	Functional	Each refueling period prior to fuel handling
10. Intake Pump House Floor (Elevation 262 ft. 6 in.)	(a) Silt Accumulation - Visual inspection of Intake Pump House Floor	Not to exceed 24 months
	(b) Silt Accumulation Measurement of Pump House Flow	Quarterly
11. Pressurizer Block Valve (RC-V2)	Functional*	Quarterly

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

4.4.4 DELETED

4.8 DELETED

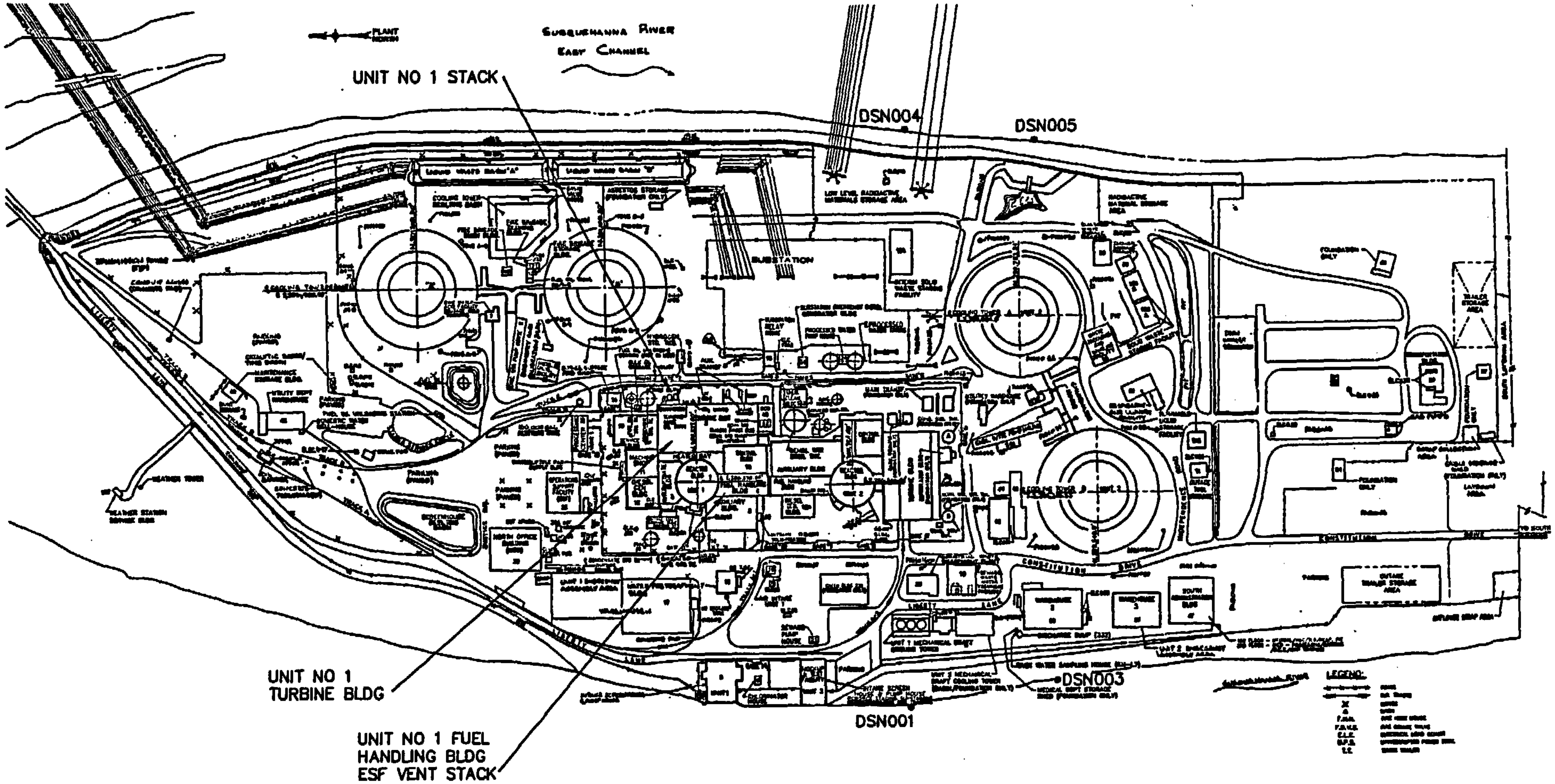


AmerGen
Three Mile Island Nuclear Station

EXTENDED PLOT PLAN

CAD FILE: 6717R1.DWG FIG 5-1

Amendment No. ~~149~~, ~~218~~, 246



1e-120-01-001 Rev. 56

Amendment No. ~~140~~; ~~210~~; 246

AmerGen
Three Mile Island Nuclear Station

Gaseous Effluent Release Point and Liquid Effluent Outlet Locations

CAD FILE: 6716R15.DWG

Fig. 5-3

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

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

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

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