ENCLOSURE 1

PAGE CHANGE LIST

PAGE CHANGE LIST

Remove the following pages from the existing Technical Specifications* and replace with new pages of the same number.

Page 4
3.2-1
3.6-1
3.11-4
3.14-1
3.14-3
3.15-1
3.22-2
3.24-1
4.1-8
4.2-2
4.4-2
4.5-1
4.6-2
5.8-1
5.9-5

Insert new page

3.2-2 Figure 3.2-1

* Includes Amendment Number 86 changes which do not become effective until July 1, 1986.

ENCLOSURE 2

DESCRIPTION OF CHANGES

TECHNICAL SPECIFICATIONS

PAGE 4

The definition of Containment Integrity has been deleted from the Definitions Section since definition appears in the actual Technical Specifications concerning containment integrity. In order to minimize confusion due to possible conflicts in the Definitions we have chosen to delete it here.

PAGE 3.2-1

The Specification has been revised following the recommendations of Generic Letter 85-19 (Reporting Requirements on Primary Coolant Iodine Spikes). As described in the Generic Letter, the immediate reporting requirements of indications of significant degradation of fuel is already incorporated in the implementation of 10 CFR 50.73 (Licensee Event Reporting System). The reporting requirement stated in our Technical Specifications have been replaced with the recommended reporting requirements and moved to the Administrative Controls section of the Technical Specifications. In addition, we are incorporating the figure from Combustion Engineering Standard Technical Specifications concerning dose equivalent Iodine 131 concentration in primary coolant vs. rated thermal power. This Figure shows acceptable and unacceptable operating regions. The current Maine Yankee Technical Specification limit for iodine spikes less than 60 microcuries/gram will continue to be in effect at or above 80% rated thermal power. The limit on the figure increases below 80% rated thermal power and has been found acceptable by the Staff for other Combustion Engineering plants on a generic basis.

PAGE 3.6-1

The change to this Specification removes the term "where appropriate" and inserts a reference to Specification 3.9, in which the requirements for automatic initiation of the ECCS system is described. This change improves the clarity of the Specification.

PAGE 3.11-4

This change to the Basis of Specification 3.11 corrects an incorrect cross-reference to another Specification.

PAGE 3.14-1

Current Maine Yankee Technical Specification 3.14B is confusing and rather inelegantly expressed. This change restates it in terms of a specific limit of 1 gallon per minute reactor coolant system leakage and a specific Remedial Action. In addition a reference to Cycle 7 operation, which is no longer appropriate, has been deleted. This change does not alter the limit or the remedial action, but merely clarifies and simplifies.

PAGE 3.14-3

The description of the concentration term, C, for secondary coolant activity has been revised to correct a misprint in current Specifications.

PAGE 3.15-1

This Specification concerning reactor power anomalies has been divided into a Specification and Remedial Action for clarity. In addition, the term steady-state concentrations has been used to distinguish brief transients from ongoing conditions.

PAGE 3.22-2

This change to the Basis of this Specification points out the applicability of the feedwater trip system to the main feedwater system.

PAGE 3.24-1

The change to the Basis of this Specification is requested in order to correct a misprint replacing 0.1 gpm primary to secondary leak rate with 1.0 gpm. This brings the Basis of this Specification into conformance with the assumptions of the FSAR.

PAGE 4.1-8

The change to Table 4.1-2 reflects the upgrade to the Refueling Water Storage Tank level instrumentation made during the 1985 refueling outage. It also clarifies the function being tested as that of the recirculation actuation signal.

PAGE 4.2-2

The term "fluoride" is added to the reactor coolant sample chemistry requirement. This is consistant with Staff guidance and current Maine Yankee practice and the requirements of Specification 3.18.

PAGES 4.2-6 and 4.2-7

The requirement to calibrate the post accident hydrogen monitor has been deleted here as it is included in Table 4.1-3. This eliminates duplication. The pages are renumbered to conform to the changes made by Amendment 86.

PAGE 4.4-2

The correction to the Basis of this Specification merely corrects a typographical error in the word "measure".

PAGE 4.5-1

Revision to the Specification for diesel generator testing brings the terminology used in Maine Yankee Specifications into conformance with those used in standard technical specifications and removes the term (the maximum expected emergency loading) which is not in common use within the industry, and replaces it with a commonly used term (continuous rating). Additionally, a specific rating of 2500 KW is included. Further, the duration of the test is decreased from two hours to one hour consistent with Standard Technical Specifications. This change is also consistant with ongoing Staff and industry efforts to assure the reliability and operability of the diesel generators while at the same time minimizing unnecessary wear which may have contributed to failures of diesel generators in the past.

PAGE 4.6-2

The changes on this page corrects a misspelling of the term "distribution" and corrects an outdated cross reference to another Specification.

PAGE 5.8-1

This Specification is revised to indicate the specific revision to Regulatory Guide 1.33 to which Maine Yankee has been and is currently committed in our NRC approved Quality Assurance Program.

PAGE 5.9-5

Special reporting requirements for containment leak rate tests have been deleted in this proposed change because reporting requirements for leak rates are included in Appendix J of 10 CFR 50. Reference in the Technical Specification is, therefore, redundant and unnecessary. An addition has been made to require, on an annual basis and under conditions set forth in Specification 3.2, reports of primary coolant iodine activity to be submitted with the Semi-annual Effluent Report submitted following January 1st of each year. This provides the format and content of the report required by Specification 3.2 and is consistent with the recommendations of Generic Letter 85-19.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATIONS

MISCELLANEOUS DEFINITIONS

Operable

A system, subsystem, train, component or device shall be operable or have operability when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

Operating

A system or component is operating if it is performing its safeguard or operating functions.

Control Element Assemblies

All full-length shutdown and regulating control element assemblies (CEA's).

Partial-Length Control Element Assemblies

Control element assemblies (CEA) that contain neutron absorbing material only in the lower quarter of their length.

]

Fire Suppression Water System

A fire suppression water system shall consist of: A water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include yard hydrant curb valves, and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

3.2 REACTOR COOLANT SYSTEM ACTIVITY

Applicability:

Applies to measured maximum activity in the reactor coolant system.

Objective:

To ensure that the reactor coolant activity does not exceed a level commensurate with the safety of the plant personnel and the public.

Specification:

A. The specific activity of the primary coolant shall be limited to less than or equal to 1.0 micro Ci/gram DOSE EQUIVALENT I-131.

Remedial Action: If the specific activity of the primary coolant is greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeds the limit line shown on Figure 3.2-1, the reactor must be made subcritical with T_{avg} less than 500°F within 6 hours (Condition 4, Transthermal).

B. The specific activity of the primary coolant shall be limited to less than or equal to 100/$\bar{E}$ micro Ci/gram.

Remedial Action: If the specific activity of the primary coolant is greater than 100/$\bar{E}$ micro Ci/gram, the reactor must be made subcritical with T_{avg} less than 500°F within 6 hours (Condition 4 Transthermal).

C. If the specific activity of the primary coolant is greater than 1.0 micro Ci/gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ micro Ci/gram, the sampling and analysis requirements of Item 1 of Table 4.2-1 shall be performed until the specific activity of the primary coolant is restored to within its limits.

BASIS:

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of the Part 100 limit following a steam generator tube rupture.

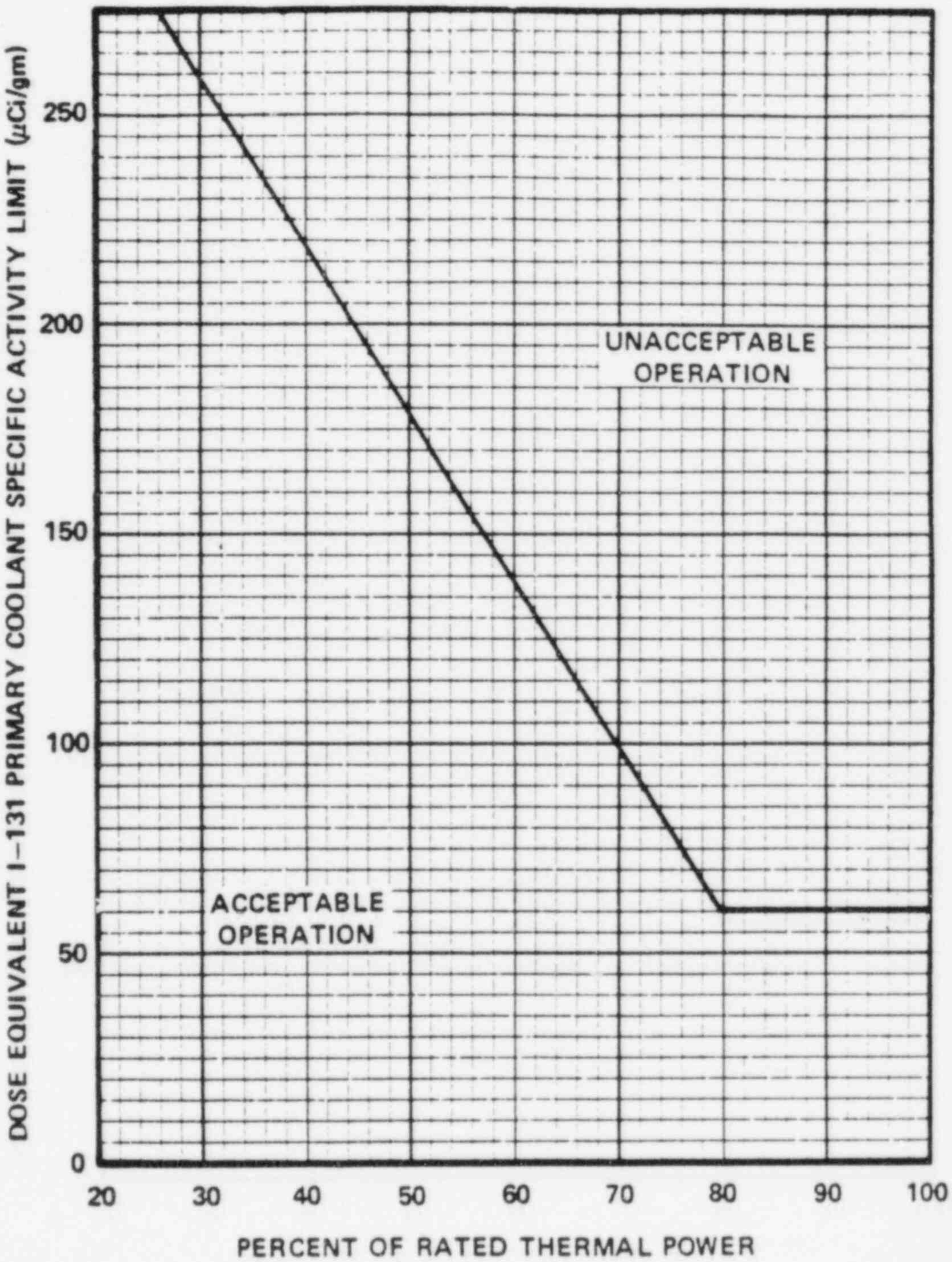

FIGURE 3.2-1

DOSE EQUIVALENT I–131 Primary Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the Primary Coolant Specific Activity $> 1.0 \mu$Ci/gram Dose Equivalent I-131

3.6 EMERGENCY CORE COOLING AND CONTANMENT SPRAY SYSTEMS

Applicability:

Applies to the operating status of the emergency core cooling and containment spray systems.

Objective:

To define the conditions under which components of the emergency core cooling and containment spray systems must be operable.

Specification:

A. The following equipment must be operable whenever the reactor coolant system temperature and pressure exceed 210°F and 400 psig:

1. Two safety injection tanks set for automatic initiation. Each tank shall contain 11,200 ± 500 gallons of water borated to at least 1720 ppm and pressurized with nitrogen to 230 psig + 10 psi, - 25 psi.

2. One operable ECCS train consisting of the following subsystems of the train. Each subsystem includes the manual valves that are aligned and locked in the position required for safeguards operation, the automatically operated valves set for automatic operation or aligned and locked in the position required for safeguards operation, the controls set for automatic initiation in accordance with Specification 3.9, and a pump powered from an engineered safeguards bus.

   a. One service water pump subsystem

   b. One component cooling pump subsystem

   c. One low pressure safety injection pump subsystem

   d. One high pressure safety injection pump subsystem

   f. One containment spray pump and RHR heat exchanger subsystem

3. Station service power in accordance with Technical Specification 3.12.A supplying the same operable ECCS train as in (2) above.

4. The refueling water storage tank and spray chemical addition tank are filled and available in accordance with Technical Specification 3.7.

5. The fill header motor operated root valves to two non-isolated loops.

Exception: The requirements may be modified with regard to the position of controls and valves during periods of hydrostatic testing.

Remedial Action: Restore required limiting condition within four hours.

Remedial Action:

If the containment weight of air monitoring system is out of service for more than ten days with the reactor critical, the Commission must be notified of plans to restore the system operability.

2. When the containment weight of air monitoring system indicates a daily air loss greater than the following, an evaluation shall be initiated to determine the validity of the indication.

   a. Equivalent to 0.15 weight percent per day at 50 psig for seven consecutive days, or

   b. Equivalent to 0.5 weight percent per day at 50 psig for four consecutive days, or

   c. Equivalent to 1.0 weight percent per day at 50 psig for three consecutive days.

3. The reactor shall be made subcritical within six hours if the evaluation required by G2:

   a. Results in identification of the source of the leak and a determination that the known containment leak rate exceeds the equivalent of 0.10 weight percent per day at 50 psig through the containment integrity boundary and cannot be isolated, or

   b. Fails to identify the source of the leakage within ten days and the containment weight of air monitoring system indication persists at an average rate in excess of 0.15 weight percent per day at 50 psig.

Basis

Specification A includes a limit of 210°F on reactor coolant temperature assures that no steam will be generated in the unlikely event of a reactor coolant system rupture and hence no driving force to release any fission products from the containment. The shutdown margins are selected based upon the types of activities that are being carried out. The higher value for refueling precludes criticality under all postulated incidents.

Specification B assures that the containment pressure boundary is defined while permitting maintenance of components necessary to integrity.
Specification 4.4 requires that the uncontrolled containment leakage conform]
to specified limits to assure that public exposure will be maintained]
well within the guidelines presented in 10 CFR 100 for the hypothetical accident described in Section 14.18 of the FSAR.

3.14 PRIMARY SYSTEM LEAKAGE

Applicability:

Applies to limiting operation of the plant under varying rates and conditions of primary system leakage.

Objective:

To specify primary plant operability with primary system leakage.

Specification:

A. When the reactor is above 2% power, two reactor coolant leak detection systems of different operating principles shall be operating, with one of the two systems sensitive to radioactivity in the containment.

Remedial Action: If two reactor coolant leakage detection systems are operable but neither is sensitive to radioactivity, a system sensitive to radioactivity must be made operable within 48 hours.

B. The reactor coolant system indicated leak rate shall be limited]
to 1 gpm or less.

Remedial Action: If the indicated leak rate should exceed 1 gpm,]
within four hours, investigate the source and assess safety implications.]

C. Reactor coolant system leakage shall not exceed any of the Specifications 1 through 5 below.

1. Leakage into the reactor containment of any magnitude that has been determined to be an indication of a deterioration of primary system pressure boundary strength welds or material.

2. Leakage into the reactor containment in excess of 1 gpm through bolted closures, valve packing, or other mechancial connections.

3. Leakage in excess of 1 gpm that is unexplained or unaccounted for.

4. Leakage in excess of 10 gpm to aerated or uncontained systems.

5. Total leakage through all steam generator tubes shall not exceed
1.0 gpm.]

Remedial Actions:

1. If the leakage specified in C.1 above has been determined to be a deterioration of primary system pressure boundary strength welds or material, then the provisions of Specification 3.0.A.2 and 3 apply.

2. If reactor coolant system leakage exceeds any of the Specifictions C.2 through C.5 above, the reactor shall be shut down within 24 hours.

]

where:

C = Secondary coolant sample activity 0.1 micro ci/cc = 0.1 Ci/M^3]

V = Water volume in three steam generator = 131 M^3 at standard conditions

B (t) = Breathing rate ($3.47 \times 10^{-4} m^3/sec$)

X/Q = 6.48×10^{-4} sec/m^3 (corresponding to Pasquill F stability and 1 m/sec wind speed)

DCF = 1.48×10^6 rem/Ci I-131 inhaled

The resulting thyroid dose is less than 1.5 rem.

3.15 REACTIVITY ANOMALIES

Applicability:

Applies to potential reactivity anomalies.

Objective:

To require evaluation of reactivity anomalies within the reactor.

Specification:

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the reactor coolant shall be periodically compared with the predicted value. The difference]
between the observed and predicted steady-state concentrations shall be]
less than the equivalent of 1% in reactivity.

Remedial Action:

The Nuclear Regulatory Commission shall be notified and an evaluation]
as to the cause of the discrepancy shall be made and reported to the Nuclear Regulatory Commission in accordance with Technical Specification 5.9.1.7.]

Basis:

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the CEA groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10% of the total core burnup. Thereafter, actual boron concentration can be compared with prediction and the reactivity status of the core can be continuously evaluated and its occurrence would be thoroughly investigated and evaluated. The methods employed in calculating the reactivity of the core vs. burnup, and the reactivity worth of boron vs. burnup, are given in the FSAR.

The system valves are aligned to provide flow to each steam generator following system actuation upon low steam generator water level signal from any one of the three steam generators. However, for a steam generator depressurization event, such as a steam line break, receipt of a low steam generator pressure signal initiates closure of the control and isolation valve(s) feeding the depressurized steam generator(s). This limits excessive reactor coolant system cooldown and the resultant reactivity insertion produced by excessive feedwater flow to a depressurized steam generator. Flow will continue to steam generators remaining pressurized. Flow to a depressurized steam generator will be reestablished by reopening the control and isolation valves after repressurization e.g., by isolation from the steam line break.

Operability of the system assures that the reactivity attributable to reactor coolant system cooldown due to feedwater addition to steam generators after a main steam line break is within the limits established in the steam line break safety analysis.

If the feedwater trip system is discovered to be inoperable, the best course of action is to restore its operability promptly, thus avoiding challanges to plant systems that result from perturbing steady state operation. A two-hour time period presents low risk of a main steam line break yet allows enough time for deliberate restoration of system operability through maintenance actions.

If operability cannot be restored the reactor must be shut down. Six hours provides ample time for an orderly controlled shutdown. If operability cannot be restored by that time, the reactor coolant system must be borated to hot shutdown concentration within an additional six hours. Twelve hours permits an orderly shutdown while assuring that the risk of a main steam line break during the period is very low.

The intended function of the feedwater trip system can be accomplished under conditions of partial system inoperability provided all main feedwater]
system pumps and valves tripped by the system which are operating can be tripped by the operable portions of the trip system. Pumps which cannot be tripped by the trip system due to partial trip system inoperability can be shut down to assure functional capability.

When the reactor coolant system is at hot shutdown boron concentration, the steam line break cooldown cannot cause sufficient reactivity insertion to cause a return to critical, so the feed trip system is not required to function.

3.24 SECONDARY COOLANT ACTIVITY

Applicability:

Applies to measured maximum activity in the secondary coolant system.

Objective:

To ensure that the secondary coolant activity does not exceed a level commensurate with the safety of the plant personnel and the public.

Specification:

The specific activity of the secondary coolant system shall be less than or equal to 0.10 micro Ci/gram DOSE EQUIVALENT I-131.

Basis:

The limitations on secondary system specific activity insure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in a steam line rupture. This dose includes that contributed by a 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line.]

Note: The secondary coolant activity surveillance requirements are given in Table 4.2-1, Item. 7.

Channel Description	Surveillance Fuctions	Frequency	Surveillance Method
8. Manual Containment Isolation Initiation	Test	R	Manual switch test
9. Manual Initiation Containment Spray	Test	R	Manual switch operation.
10. Refueling Water Tank Level RAS Initiation	Test	R	Fluid removed from level transmitters to verify actuation of valves.]]
11. Safety Injection Tanks Level and Pressure	a. Check	S (3)	a. Verify level and pressure.
	b. Calibrate	R	b. Known pressure and differential pressure applied to pressure and level sensors.
12. Main Steam Isolation Valve Circuits	a. Check	S (3)	a. Compare four independent steam generator pressure indications.
	b. Calibrate	R	b. Simulated signal applied to motor relays to verify trip points, logic operation, solenoid valve operation
13. High Pressure Safety Injection Header Pressure	a. Check	M (3)(4)	a. Verify header pressure indication during pump test.
	b. Calibrate	R	b. Known pressure applied to sensors.
14. Low Pressure Safety Injection Header Pressure	a. Check	M (3)(4)	a. Verify header pressure indication during pump test.
	b. Calibrate	R	b. Known pressure applied to sensors.

Table 4.2-1

Minimum Frequencies for Sampling Tests

	Test	Frequency
1. Reactor Coolant Samples	Gross Activity Determination	At least once per 72 hours.
	Isotopic Analysis for DOSE EQUIVALENT I-131 concentration	1 per 14 days
	Radiochemical for $\overline{E}$ Determination	1 per 6 months(a)
	Isotopic Analysis for Iodine, including I-131, I-133, and I-135.	Until the specific activity of the primary coolant system is restored within the limits, once per 4 hours, whenever the DOSE EQUIVALENT I-131 exceeds 1.0 uCi/gram, and One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one-hour period.
	Chemistry (chloride and fluoride)	3 times per week]
	Hydrogen	Weekly(d)
	Oxygen	Prior to Heatup above 250°F, and 3 times per week when above 250°F.
	Process Radiation Monitor	Continuous(b)(d)
2. Reactor Coolant Boron	Boron Concentration	3 times per week
	Boronometer	Continuous(b)(d)
3. Refueling Water Tank Water Sample	Boron Concentration	Monthly(e)(g)

Table 4.2-2

Minimum Frequencies for Equipment Tests

	Test	Frequency
1. Control Element Assemblies (CEAs)	Drop Times of all full-length CEAs	Each refueling interval
2. Control Element Assemblies	Partial Movement of all CEAs (minimum of 6")	Every two weeks when the reactor is critical
3. Pressurizer Safety Valves	Set Point	One valve each refueling interval
4. Main Steam Safety Valves	Set Point	2 valves per steam generator, each refueling interval
5. Refueling System Interlocks	Functioning	Prior to refueling operations
6. Primary system Leakage	Evaluate	Daily**
7. Diesel Fuel Supply	Fuel Inventory	Weekly
8. Deleted		
9. Turbine stop governor, Reheater and Intercept Valves	Functioning	Monthly when the turbine is operating
10. L.P. Turbine Rotor Inspection	Visual, Magnetic Particle or Liquid Penetrant	One Rotor each 4 years
11. Post-Accident Containment Vent System		
a. System valves	Verify operability	Within one month of startup from each refueling shutdown.
b. Flowpath	Verify system flow capability by observing flow indication on the system flowmeter	Within one month of startup from each refueling shutdown.

Table 4.2-2 (Continued)

Minimum Frequencies for Equipment Tests

	Test	Frequency
12. PORV and PORV Block Valve operability test		
a. Block Valves	Verify operability by operating the valve(s) through one complete cycle of full travel	At least once per 92 days.
b. PORVs	Verify operability by manual actuation of the control circuitry	At least once per 18 months.
13. Pressurizer	Verify Level***	At least once per 12 hours.

* Filters for containment and fuel storage building purging

** Whenever the reactor coolant system is at or above operating pressure

*** Only required when reactor coolant system T_{ave} is greater than 500°F.

2. Type C containment leakage rate tests will be performed in accordance with the provisions of 10 CFR 50.54(o) and 10 CFR 50 Appendix J.

   This requirement is effective following the Cycle 8/9 refueling outage. It will be implemented as practicable prior to that time.

3. The Type B air lock tests will be performed in accordance with the provisions in 10 CFR 50.54(o) and Appendix J, at a test pressure of not less than 50 psig. The allowable leak rate for the air lock test is 30 lbm per day or $0.05L_a$.

4. Containment purge supply, exhaust and bypass valve testing requirements for on line purging are as follows:

   a) The containment purge supply, exhaust and bypass valves will be leak tested within 24 hours following the termination of on-line purge, except when the valves are being used for multiple cyclings, then at least once per 72 hours.

   b) The containment purge supply, exhaust and bypass valves will be leak tested at six month intervals.

Basis

A leakage rate value of 0.10 weight percent per 24 hours will, under the most adverse design basis accident conditions, maintain public exposure well below 10 CFR 100 values in the event of the hypothetical accident. The tests at reduced pressure will assure the continued ability of the containment to perform its function.

The air lock leak test is a measure of the operability of the air lock, however, the air lock is still considered to be operable in the event of exceeding air lock leak test acceptance criteria as long as one hatch is determined to be properly closed and sealed in accordance with Technical Specification 3.11 and total containment leakage from all penetrations and isolation valves (including the air lock) has been evaluated to be within L_p.]

L_p in pounds mass per day = (0.6)(0.1%) (609,000 lbm) = 365 lbm per 24 hours at 50 psig.

The seal is determined to be operable by an appropriate pressure test.

References: FSAR Sections 5 and 14.18
10 CFR 50.54, Paragraph (o), and 10 CFR 50, Appendix J, Reactor Containment Testing Requirements

4.5 EMERGENCY POWER SYSTEM PERIODIC TESTING

Applicability: Applies to the periodic testing requirements of the station emergency electrical power systems.

Objective: To verify the operability of the station emergency electrical power systems.

Specification: A. Diesel Generators:

The following tests shall be performed:

1. Manually initiated demonstration of the ability of each diesel generator to start and deliver power up to its continuous rating (2500 kW)] when operating in parallel with other power sources. This test will be conducted monthly whenever Plant Conditions are as defined in Section 3.6.A of these Specifications, and shall be of at least one hour] duration, and include operation of the fuel oil transfer pumps.

2. Demonstration of the readiness of the diesel generator to start automatically and restore power to vital equipment by initiating or simulating loss of all normal a-c station service power supplies. This test will be conducted during each refueling interval.

B. Station Batteries:

Each week the specific gravity and voltage of the pilot cell of each of the two main station batteries that are associated with the d-c buses feeding the safeguards equipment shall be measured and an overall visual inspection of each battery shall be performed. Every other month the liquid level, the specific gravity and the voltage of all cells of each of these batteries whall be checked; and during the initial refueling interval, and every third refueling interval thereafter, they shall be subjected to a rated load discharge test.

Basis: The test of the diesel generators is conducted to demonstrate that the diesel generators will provide adequate power for operation of vital equipment.

The test of the diesel generators during each refueling interval will functionally test automatic diesel starting, closure of diesel breaker and emergency bus loading and load shedding. The diesel generator will be started by initiating or simulating loss of normal a-c station service power.

One LPSI pump and one CS pump shall be flow tested at 100 psi discharge head.

During these tests flow distribution through the HPSI and LPSI flow orifices will be checked.]

Acceptance performance shall be that the pumps and orifices attain flow values used in the safety analysis.

Alternate pumps will be tested at each refueling interval, so that all pumps will be tested within any five year period.

b. ECCS Valves:

All automatically operated valves and the motor operated fill header root valves shall be exercised through their full travel in conjunction with the actuation signal testing set forth in Table 4.1-2 of Technical Specifications.

c. Safety Injection Tanks:

Each safety injection tank will be flow tested by opening the tank isolation valve sufficient to verify check valve operation.

d. The correct position of each electrical and mechanical position stop for the following throttle valves shall be verified:

1) Within 4 hours following completion of maintenance on the valve when the HPSI system is required to be operable.

2) At least once per 4 months

<u>Valve Numbers</u>

HSI-M-11
HSI-M-12
HSI-M-21
HSI-M-22
HSI-M-31
HSI-M-32

e. A flow balance test, as described in 4.6.A.2 above, shall be performed during shutdown to confirm the injection flow rates assumed in the Safety Analysis following completion of HPSI or LPSI system modifications that alter system flow characteristics.

f. ECCS Check Valves

The check valve barriers defined in Technical Specification 3.19.A.4 shall be determined to be intact by leak testing.]

5.8 PROCEDURES

5.8.1 Written procedures shall be established, implemented and maintained coveri g the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, (Rev. 2), February, 1978.]

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

5.8.2 Each procedure of 5.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

5.8.3 Temporary changes to procedures of 5.8.1 above may be made provided:

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

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License.

c. The change is documented, reviewed by the PORC and approved by the Plant Manager within 14 days of implementation.

SPECIAL REPORTS

5.9.1.7 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report. These reports shall be submitted concerning the activities identified below pursuant to the requirements of the applicable specification or rule.

a. Reactivity anomalies, Specification 3.15.

b. Excessive radioactive release, Specifications 3.16A2 and 3.17A2.

c. Plans for restoration of 115 kV service, Specification 3.12.

d. Total dose Specification 3.16 and 3.17.

e. Primary Coolant Iodine Activity.

If specific activity of the primary coolant exceeded the limits of Specification 3.2., a report shall be filed with the Semi-Annual Report following January 1 of each year. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.