b. If an unscheduled loss of one or more reactor coolant pumps occurs while operating below 10% RATED POWER (P-7) and results in less than two pumps in service, the affected plant shall be shutdown and the reactor made subcritical by inserting all control banks into the core. The shutdown rods may remain withdrawn.

c. When the average reactor coolant loop temperature is greater than 350°F, the following conditions shall be met:

    1. At least two reactor coolant loops shall be OPERABLE.

    2. At least one reactor coolant loop shall be in operation.

d. When the average reactor coolant loop temperature is less than or equal to 350°F, the following conditions shall be met:

    1. A minimum of two non-isolated loops, consisting of any combination of reactor coolant loops or residual heat removal loops, shall be OPERABLE, except as specified below:

        (a) One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

        (b) During REFUELING OPERATIONS the residual heat removal loop may be removed from operation as specified in TS 3.10.A.4.

    2. At least one reactor coolant loop or one residual heat removal loop shall be in operation, except as specified in Specification 3.10.A.4.

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize to 0.5 psig within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig for the interval from 1 to 4 hours following the Design Basis Accident.

If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the HOT SHUTDOWN condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

UFSAR Section 4.2.2.4	Reactor Coolant Pump
UFSAR Section 5.2	Containment Isolation
UFSAR Section 5.2.1	Design Bases
UFSAR Section 5.2.2	Isolation Design
UFSAR Section 5.3.4	Containment Vacuum System

During refueling, the reactor refueling water cavity is filled with approximately 220,000 gal of water borated to at least 2,300 ppm boron. The boron concentration of this water, established by Specification 3.10.A.7, is sufficient to maintain the reactor subcritical by at least 5% Δk/k in the COLD SHUTDOWN condition with all control rod assemblies inserted. This includes a 1% Δk/k and a 50 ppm boron concentration allowance for uncertainty. This concentration is also sufficient to maintain the core subcritical with no control rod assemblies inserted into the reactor. Checks are performed during the reload design and safety analysis process to ensure the K-effective is equal to or less than 0.95 for each core. Periodic checks of refueling water boron concentration assure the proper shutdown margin. Specification 3.10.A.8 allows the Control Room Operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are used during refueling to assure safe handling of the fuel assemblies. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

b. Before opening the hot leg loop stop valve.

1) The boron concentration of the isolated loop shall be greater than or equal to the boron concentration corresponding to the shutdown margin requirements of Specification 1.0.C.2 or 3.10.A.7, as applicable for the active volume of the Reactor Coolant System. Verification of this condition shall be completed within 1 hour prior to opening the hot leg stop valve in the isolated loop.

c. Before opening the cold leg loop stop valve.

1) The hot leg loop stop valve shall be open with relief line flow established for at least 90 minutes at greater than or equal to 125 gpm.

2) The cold leg temperature of the isolated loop shall be at least 70°F and within 20°F of the highest cold leg temperature of the active loops. Verification of this condition shall be completed within 30 minutes prior to opening the cold leg stop valve in the isolated loop.

3) The boron concentration of the isolated loop shall be greater than or equal to the boron concentration corresponding to the shutdown margin requirements of Specification 1.0.C.2 or 3.10.A.7, as applicable for the active volume of the Reactor Coolant System. Verification of this condition shall be completed after relief line flow for at least 90 minutes at greater than or equal to 125 gpm and within 1 hour prior to opening the cold leg stop valve in the isolated loop.

5. Whenever an isolated and drained reactor coolant loop is filled from the active volume of the RCS, the following conditions shall apply:

a. Seal injection may be initiated to the reactor coolant pump in the isolated loop provided that:

1) The isolated loop is drained. Verification of this condition shall be completed within 2 hours prior to initiating seal injection.

2) The boron concentration of the source for reactor coolant pump seal injection shall be greater than or equal to the boron concentration corresponding to the shutdown margin requirements of Specification 1.0.C.2 or 3.10.A.7, as applicable for the active volume of the Reactor Coolant System. If using the Volume Control Tank (VCT) as the source for reactor coolant pump seal injection, verification of the boron concentration shall be completed within 1 hour prior to initiating seal injection and every hour thereafter during the loop backfill evolution.

b. The cold leg loop stop valve may be energized and/or opened to backfill the loop from the active volume of the Reactor Coolant System provided that:

1) The isolated loop is drained or reactor coolant pump seal injection has been initiated in accordance with Specification 3.17.5.a above. Verification of the loop being drained shall be completed within 2 hours prior to partially opening the cold leg stop valve in the isolated loop.

2) The Reactor Coolant System level is at least 18 ft.

3) A source range nuclear instrumentation channel is OPERABLE with audible indication in the control room.

c. Backfilling of the isolated loop may continue provided that:

1) The Reactor Coolant System level is maintained at or above 18 ft. If Reactor Coolant System level is not maintained at or above 18 ft. the loop stop valve shall be closed.

2) The boron concentration of the reactor coolant pump seal injection source is greater than or equal to the boron concentration corresponding to the shutdown margin requirements of Specification 1.0.C.2 or 3.10.A.7, as applicable for the active volume of the Reactor Coolant System. If the boron concentration is not maintained greater than or equal to the required boron concentration noted above, the loop stop valve on the loop being backfilled shall be closed and either drain the loop or apply Specification 3.17.4.

3) A source range nuclear instrumentation channel is OPERABLE and continuously monitored with audible indication in the control room during the backfill evolution. Should the count rate increase by more than a factor of two over the initial count rate, the cold leg loop stop valve shall be closed and no attempt made to open the cold leg stop valve until the reason for the count rate increase has been determined.

d. When the isolated loop is full, the cold leg loop stop valve can be fully opened and the hot leg loop stop valve opened provided that:

1) The boron concentration of the isolated loop is greater than or equal to the boron concentration corresponding to the shutdown margin requirements of Specification 1.0.C.2 or 3.10.A.7, as applicable for the active volume of the Reactor Coolant System. If the VCT was used as the source for reactor coolant pump seal injection, this condition shall be verified within 1 hour prior to fully opening the loop stop valves. If the boron concentration in the isolated loop does not meet the condition above, close the loop stop valve and either drain the loop or apply Specification 3.17.4.

2) The hot and cold leg loop stop valves are opened within 2 hours after the isolated loop is filled. If the loop stop valves are not fully open within 2 hours, close the loop stop valves and either drain the loop or apply Specification 3.17.4.

Basis

The Reactor Coolant System may be operated with isolated loops in COLD SHUTDOWN or REFUELING SHUTDOWN in order to perform maintenance. A loop stop valve in any loop can be closed for up to two hours without restriction for testing or maintenance in these operating conditions. While operating with a loop isolated, AC power is removed from the loop stop valves and their breakers locked opened to prevent inadvertent opening. When the isolated loop is returned to service, the coolant in the isolated loop

4.0.5 Surveillance requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(f), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(f)(6)(i). Inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the surveillance requirement within the allowable outage time limits of the Action Statement requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the operability requirements of a Limiting Condition for Operation. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B), unless it meets an exception listed therein, because it is a condition prohibited by the plant's Technical Specifications.

If the allowable outage time limits of the Action Statement requirements are less than 24 hours or a shutdown is required to comply with Action Statement requirements, e.g., Specification 3.0.1, a 24 hour allowance is provided to permit a delay in implementing the Action Statement requirements. This provides an adequate time limit to complete surveillance requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with Action Statement requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. This

TABLE 4.1-1(Continued)

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

Channel Description	Check	Calibrate	Test	Remarks
32. Auxiliary Feedwater				
a. Steam Generator Water Level Low-Low	S	R	Q(1)	1) The auto start of the turbine driven pump is not included in the quarterly test, but is tested within 31 days prior to each startup.
b. RCP Undervoltage	S	R	R(1)(2)	1) The actuation logic and relays are tested within 31 days prior to each startup. 2) Setpoint verification not required.
c. S.I.	(All Safety Injection surveillance requirements)			
d. Station Blackout	N.A.	R	N.A.	
e. Main Feedwater Pump Trip	N.A.	N.A.	R	
33. Loss of Power				
a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	Q(1)	1) Setpoint verification not required.
b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	Q(1)	1) Setpoint verification not required.
34. Deleted				
35. Manual Reactor Trip	N.A.	N.A.	R	The test shall independently verify the operability of the undervoltage and shunt trip attachments for the manual reactor trip function. The test shall also verify the operability of the bypass breaker trip circuit.
36. Reactor Trip Bypass Breaker	N.A.	N.A.	M(1), R(2)	1) Remote manual undervoltage trip immediately after placing the bypass breaker into service, but prior to commencing reactor trip system testing or required maintenance. 2) Automatic undervoltage trip.
37. Safety Injection Input to RPS	N.A.	N.A.	R	
38. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	

TABLE 4.1-2A

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
1. Control Rod Assemblies	Rod drop times of all full length rods at hot conditions	Prior to reactor criticality: a. For all rods following each removal of the reactor vessel head b. For specially affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and c. Once per 18 months	7
2. Control Rod Assemblies	Partial movement of all rods	Quarterly	7
3. Refueling Water Chemical Addition Tank	Functional	Once per 18 months	6
4. Pressurizer Safety Valves	Setpoint	Per TS 4.0.5	4
5. Main Steam Safety Valves	Setpoint	Per TS 4.0.5	10
6. Containment Isolation Trip	* Functional	Once per 18 months	5
7. Refueling System Interlocks	* Functional	Prior to refueling	9.12
8. Service Water System	* Functional	Once per 18 months	9.9
9. Deleted			
10. Primary System Leakage	* Evaluate	Daily	4
11. Diesel Fuel Supply	* Fuel Inventory	5 days/week	8.5
12. Deleted			
13. Main Steam Line Trip Valves	Functional (Full Closure)	Before each startup (TS 4.7) The provisions of Specification 4.0.4. are not applicable	10

The containment is designed for a maximum pressure of 45 psig. The containment is maintained at a subatmospheric air partial pressure consistent with TS Figure 3.8-1 depending upon the cooldown capability of the Engineered Safeguards and will not rise above 45 psig for any postulated loss-of-coolant accident.

The initial test pressure for the Type A test is 47.0 psig to allow for containment expansion and equalization. A review was performed to determine the effects of pressurizing containment above its design pressure of 45.0 psig. This review was based on the original containment test at 52 psig. During that test, the calculated stresses were found to be well within the allowable yield strength of the structural reinforcing bars, therefore performance of the Type A test at 47 psig will have no detrimental effect on the containment structure.

All loss-of-coolant accident evaluations have been based on an integrated containment leakage rate not to exceed 0.1% of containment volume per 24 hr.

The above specification satisfies the conditions of 10 CFR 50.54(o) which stated that primary reactor containments shall meet the containment leakage test requirements set forth in Appendix J.

The limitations on closure and leak rate for the containment airlocks are required to meet the restrictions on containment integrity and containment leak rate. Surveillance testing of the airlock seals provides assurance that the overall airlock leakage will not become excessive due to seal damage during the intervals between airlock leakage tests.

References

UFSAR Section 5.5	Containment Tests and Inspections
UFSAR Section 7.5.1	Design Bases of Engineered Safeguards Instrumentation
UFSAR Section 14.5	Loss of Coolant Accident
10 CFR 50 Appendix J	"Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors"

f. Responsibilities

The SNSOC shall be responsible for:

1. Review of a) all new normal, abnormal, and emergency operating procedures and all new maintenance procedures, b) all procedure changes that require a regulatory evaluation, and c) any other procedures or changes thereto as determined by the Site Vice President which affect nuclear safety.

2. Review of all new test and experiment procedures that affect nuclear safety.

3. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

4. Review of proposed changes to Technical Specifications and shall submit recommended changes to the Site Vice President.

5. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President - Nuclear Operations and to the Management Safety Review Committee.

6. Review of all Reportable Events and special reports submitted to the NRC.

7. Review of facility operations to detect potential nuclear safety hazards.

8. Performance of special reviews, investigations or analyses and report thereon as requested by the Chairman of the SNSOC or Site Vice President.

9. Deleted.

10. Deleted.

11. Review of every unplanned onsite release of radioactive material to the environs exceeding the limits of Specification 3.11, including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President - Nuclear Operations and to the Management Safety Review Committee.

12. Review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

13. Review of the Fire Protection Program and implementing procedures and shall submit recommended Program changes to the designated offsite management responsible for reviewing changes that pertain to Fire Protection.

g. Authority

The SNSOC shall:

1. Provide written approval or disapproval of items considered under (1) through (3) above. SNSOC approval shall be certified in writing by either the Manager - Station Operations and Maintenance or the Manager - Station Safety and Licensing.

2. Render determinations in writing with regard to whether or not each item considered under (1) through (5) above requires a license amendment request.

3. Provide written notification within 24 hours to the Vice President - Nuclear Operations and to the Management Safety Review Committee of disagreement between SNSOC and the Site Vice President; however, the Site Vice President shall have responsibility for resolution of such disagreements pursuant to 6.1.A above.

h. Records

The SNSOC shall maintain written minutes of each meeting and copies shall be provided to the Vice President - Nuclear Operations and to the Management Safety Review Committee.

e. Meeting Frequency

The MSRC shall meet at least once per calendar quarter.

f. Quorum

The minimum quorum of the MSRC necessary for the performance of the MSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 50% of the MSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

g. Review

The MSRC shall be responsible for the review of:

1. Regulatory reviews as programmatically discussed in the Updated Final Safety Analysis Report for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to assess the effectiveness of the safety and regulatory review program and to verify it is effective in identifying changes that require a license amendment pursuant to Section 50.59, 10 CFR.

2. Proposed changes to procedures, equipment or systems which require a license amendment as defined in Section 50.59, 10 CFR.

3. Proposed tests or experiments which require a license amendment as defined in Section 50.59, 10 CFR.

4. Proposed changes to Technical Specifications or the Operating Licenses.

2. The requirements of 6.4.B.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr, but less than 500 rads/hr at one meter from a radiation source or any surface through which radiation penetrates. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty and/or the senior station individual assigned the responsibility for health physics and radiation protection.

3. Written procedures shall be established, implemented, and maintained covering the activities referenced below:

    a. Process Control Program implementation.

    b. Offsite Dose Calculation Manual implementation.

C. All procedures described in 6.4.A and 6.4.B shall be reviewed and approved by the Station Nuclear Safety and Operating Committee (SNSOC) prior to implementation. Subsequent procedure changes that require a regulatory evaluation shall also be reviewed and approved by SNSOC prior to implementation. All other changes shall be independently reviewed and approved as discussed in the Updated Final Safety Analysis Report.

L. <u>Iodine Monitoring</u>

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital area under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

M. Deleted

resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operations), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

2. Annual Reports[1]

   a. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions[2], e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

Note: Footnotes 1 and 2 are located on page TS 6.6-11.

6.8 PROCESS CONTROL PROGRAM AND OFFSITE DOSE CALCULATION MANUAL

A. Process Control Program (PCP)

Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program Topical Report. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall require review and acceptance by the SNSOC and the approval of the Site Vice President prior to implementation.

B. Offsite Dose Calculation Manual (ODCM)

Changes to the ODCM:

1. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program Topical Report. This documentation shall contain:

   a. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and