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

September 29, 1981

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Posted
Amdt. 73
to DPR-37

Docket Nos. ~~50-280~~
and 50-281

Mr. R. H. Leasburg
Vice President, Nuclear Operations
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

Dear Mr. Leasburg:

The Commission has issued the enclosed Amendment No. 72 to Facility Operating License No. DPR-32 and Amendment No. 73 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 14, 1980, as supplemented December 23, 1980 and August 21, 1981. The Technical Specifications changes are supported by the Safety Evaluation Report as transmitted to the Virginia Electric and Power Company by letter dated April 24, 1980.

These amendments incorporate the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. They specifically include the areas of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation, auxiliary feedwater systems, and the implementation of programs to reduce leakage outside containment and to accurately determine airborne iodine concentrations.

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

A copy of the Notice of Issuance is also enclosed.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 72 to DPR-32
2. Amendment No. 73 to DPR-37
3. Notice of Issuance

cc w/enclosures:
See next page

Mr. R. H. Leasburg
Virginia Electric and Power Company

cc: Mr. Michael W. Maupin
Hunton and Williams
Post Office Box 1535
Richmond, Virginia 23213

Mr. J. L. Wilson, Manager
P. O. Box 315
Surry, Virginia 23883

Swem Library
College of William and Mary
Williamsburg, Virginia 23185

Donald J. Burke, Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
Post Office Box 166
Route 1
Surry, Virginia 23883

Mr. Sherlock Holmes, Chairman
Board of Supervisors of Surry County
Surry County Courthouse, Virginia 23683

Commonwealth of Virginia
Council on the Environment
903 Ninth Street Office Building
Richmond, Virginia 23219

Attorney General
1101 East Broad Street
Richmond, Virginia 23219

Mr. James R. Wittine
Commonwealth of Virginia
State Corporation Commission
Post Office Box 1197
Richmond, Virginia 23209

Regional Radiation Representative
EPA Region III
Curtis Building - 6th Floor
6th and Walnut Streets
Philadelphia, Pennsylvania 19106

Mr. J. H. Ferguson
Executive Vice President - Power
Virginia Electric and Power Company
Post Office Box 26666
Richmond, Virginia 23261

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 72
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 14, 1980, as supplemented December 23, 1980 and August 21, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 72, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 29, 1981

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

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 73
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 14, 1980, as supplemented December 23, 1980 and August 21, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 73, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 29, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 72 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.1-1	3.1-1
3.1-2	3.1-2
3.1-2a	--
3.1-3	3.1-3
3.1-4	3.1-4
3.1-5	3.1-5
--	3.1-5a
--	3.1-5b
--	3.1-5c
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-5	3.7-5
3.7-6	3.7-6
3.7-7	3.7-7
3.7-8	3.7-8
3.7-9	3.7-9
3.7-10	3.7-10
3.7-11	3.7-11
3.7-12	3.7-12
3.7-13	3.7-13
3.7-14	3.7-14
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--	3.7-20
--	3.7-21
3.8-1	3.8-1
3.8-2	3.8-2
3.8-3	3.8-3
3.8-4	3.8-4
--	3.8-5
--	3.8-6
--	3.8-7
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--	3.8-9
--	3.8-10
--	3.8-11

Remove Pages	Insert Pages
--	3.8-12
--	3.8-13
--	3.8-14
--	3.8-15
--	3.8-16
--	3.8-17
--	3.8-18
--	3.8-19
--	3.8-20
4.1-1	4.1-1
--	4.1-1a
4.1-6	4.1-6
4.1-7	4.1-7
4.1-8	4.1-8
4.1-9	4.1-9a
4.1-9a	4.1-9b
--	4.1-9c
--	4.1-9d
6.4-6	6.4-6
--	6.4-7
6.6-4	6.6-4

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objectives

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

These conditions relate to: operational components, heatup and cooldown, leakage, reactor coolant activity, oxygen and chloride concentrations, minimum temperature for criticality, and reactor coolant system overpressure mitigation.

A. Operational Components

Specifications

1. Reactor Coolant Pumps

   a. A reactor shall not be brought critical with less than two pumps, in non-isolated loops, in operation.

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 in Specification 3.10.A.6.

   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.6.

e. Reactor power shall not exceed 50% of rated power with only two pumps in operation unless the overtemperature ΔT trip setpoints have been changed in accordance with Section 2.3, after which power shall not exceed 60% with the inactive loop stop valves open and 65% with the inactive loop stop valves closed.

f. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loop shall be operable when the average reactor coolant temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. One valve shall be operable whenever the head is on the reactor vessel, except during hydrostatic tests.

b. Three valves shall be operable when the reactor coolant average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

c. Valve lift settings shall be maintained at 2485 psig ± 1 percent.

4. Reactor Coolant Loops

Loop stop valves shall not be closed in more than one loop unless the Reactor Coolant System is connected to the Residual Heal Removal System and the Residual Heat Removal System is operable.

5. Pressurizer

a. The reactor shall be maintained subcritical by at least 1% until the steam bubble is established and necessary sprays and at least 125 Kw of heaters are operable.

b. With the pressurizer inoperable due to inoperable pressurizer heaters, restore the inoperable heaters within 72 hours or be in at least hot shutdown within 6 hours and the reactor coolant system temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

c. With the pressurizer otherwise inoperable, be in at least hot shutdown with the reactor trip breakers open within 6 hours and the reactor coolant system temperature and pressure less than 350°F and 450 psig, respectively, within the following 12 hours.

6. Relief Valves

a. Two power operated relief valves (PORVs) and their associated block valves shall be operable whenever the reactor keff is ≧0.99.

b. With one or more PORVs inoperable, within 1 hour either restore the PORV(s) to operable status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

c. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to operable status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Basis

Specification 3.1.A-1 requires that a sufficient number of reactor coolant pumps be operating to provide coastdown core cooling flow in the event of a loss of reactor coolant flow accident. This provided flow will maintain the

DNBR above 1.30.[1] Heat transfer analyses also show that reactor heat equivalent to approximately 10% of rated power can be removed with natural circulation; however, the plant is not designed for critical operation with natural circulation or one loop operation and will not be operated under these conditions.

When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the equivalent of the reactor coolant system volume in approximately one half hour.

One steam generator capable of performing its heat transfer function will provide sufficient heat removal capability to remove core decay heat after a normal reactor shutdown. The requirement for redundant coolant loops ensures the capability to remove core decay heat when the reactor coolant system average temperature is less than or equal to 350°F. Because of the low-low steam generator water level reactor trip, normal reactor criticality cannot be achieved without water in the steam generators in reactor coolant loops with open loop stop valves. The requirement for two operable steam generators, combined with the requirements of Specification 3.6, ensure adequate heat removal capabilities for reactor coolant system temperatures of greater than 350°F.

Each of the pressurizer safety valves is designed to relieve 295,000 lbs. per hr. of saturated steam at the valve setpoint. Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. There are no credible accidents which could occur when the Reactor Coolant System is connected to the Residual Heat Removal System which could give a surge rate exceeding the capacity of one pressurizer safety valve. Also, two safety valves have a capacity greater than the maximum surge rate resulting from complete loss of load.[2]

The limitation specified in item 4 above on reactor coolant loop isolation will prevent an accidental isolation of all the loops which would eliminate the capability of dissipating core decay heat when the Reactor Coolant System is not connected to the Residual Heat Removal System.

The requirement for steam bubble formation in the pressurizer when the reactor has passes 1% subcriticality will ensure that the Reactor Coolant System will not be solid when criticality is achieved.

The requirement that 125 Kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at hot shutdown.

The power operated relief valves (PORVs) operate to relieve RCS pressure below the setting of the pressurizer code safety valves. These relief valves have remotely operated block valves to provide a positive shutoff capability should a relief valve become inoperable. The electrical power for both the relief valves and the block valves is capable of being supplied from an emergency power source to ensure the ability to seal this possible RCS leakage path.

References:

(1) FSAR Section 14.2.9

(2) FSAR Section 14.2.10

3.7 INSTRUMENTATION SYSTEMS

Operational Safety Instrumentation

Applicability:

Applies to reactor and safety features instrumentation systems.

Objectives:

To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

A. For on-line testng or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with TS Tables 3.7-1 through 3.7-3.

B. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, or Minimum Degree of Redundancy cannot be achieved, operation shall be limited according to the requirement shown in Column 4 of TS tables 3.7-1 through 3.7-3.

C. In the event of sub-system instrumentation channel failure permitted by Specification 3.7-B, Tables 3.7-1 through 3.7-3 need not be observed during the short period of time and operable sub-system channel are tested where the failed channel must be blocked to prevent unnecessary reactor trip.

D. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in TS Table 3.7-4.

E. Automatic functions operated from radiation monitor alarm shall be as stated in TS Table 3.7-5. The requirements of Specification 3.0.1 are not applicable.

F. The accident monitoring instrumentation for its associated operable components listed in TS Table 3.7-6 shall be operable in accordance with the following:

1. With the number of operable accident monitoring instrumentation channels less than the total number of channels shown in TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 7 days or be in at least hot shutdown within the next 12 hours.

2. With the number of operable accident monitoring instrumentaton channels less than the minimum channels operable requirement of TS Table 3.7-6, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

Basis

Instrument Operating Conditions

During plant operations, the complete instrumentation system will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The nuclear instrumentation system channels are not intentionally placed in a tripped mode since the test signal is superimposed on the normal detector signal to test at power. Testing of the NIS power range channel requires: (a) bypassing the Dropped Rod protection from NIS, for the channel being tested: and (b) placing the $\Delta T/T_{avg}$ protection channel set that is being fed from the NIS channel in the trip mode and (c) defeating the power mismatch section of T_{avg} control channels when the appropriate NIS channel is being tested. However, the Rod Position System and remaining NIS channels still provide the dropped-rod protection. Testing does not trip the system unless a trip condition exists in a concurrent channel.

Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features. (1)

Safety Injection System Actuation

Protection against a Loss of Coolant or Steam Break Accident is brought about by automatic actuation of the Safety Injection System which provides emergency cooling and reduction of reactivity.

The Loss of Coolant Accident is characterized by depressurization of the Reactor Coolant System and rapid loss of reactor coolant to the containment. The Engineered Safeguards Instrumentaton has been designed to sense these effects of the Loss of Coolant accident by detecting low pressurizer pressure to generator signals actuating the SIS active phase. The SIS active phase is also actuated by a high containment pressure signal brought about by loss of high enthalpy coolant to the containment. This actuation signal acts as a backup to the low pressurizer pressure actuation of the SIS and also adds diversity to protection against loss of coolant.

Signals are also provided to actuate the SIS upon sensing the effects of a steam line break accident. Therefore, SIS actuation following a steam line break is designed to occur upon sensing high differential steam pressure between the steam header and steam generator line or upon sensing high steam line flow in coincidence with low reactor coolant average temperature or low steam line pressure.

The increase in the extraction of RCS heat following a steam line break results in reactor coolant temperature and pressure reduction. For this reason protection against a steam line brea accident is also provided by low pressurizer pressure actuating safety injection.

Protection is also provided for a steam line break in the containment by actuation of SIS upon sensing high containment pressure.

SIS actuation injects highly borated fluid into the Reactor Coolant System in order to counter the reactivity insertion brough about by cooldown of the reactor coolant which occurs during a steam line break accident.

Containment Spray

The Engineered Safety Features also initiate containment spray upon sensing a high-high containment pressure signal. The containment spray acts to reduce containment pressure in the event of a loss of coolant or steam line break accident inside the containment. The containment spray cools the containment directly and limits the release of fission products by absorbing iodine should it be released to the containment.

Containment spray is designed to be actuated at a higher containment pressure (approximately 50% of design containment pressure) than the SIS (10% of design). Since spurious actuation of containment spray is to be avoided, it is initiated only on coincidence of high-high containment pressure sensed by 3 out of the 4 containment pressure signals provided for its actuation.

Steam Line Isolation

Steam line isolation signals are initiated by the Engineered Safety Features closing all steam line trip valves. In the event of a steam line break, this action prevents continuous, uncontrolled steam release from more than one steam generator by isolating the steam lines on high-high containment pressure or high steam line flow with coincident low steam line pressure or low reactor coolant average temperature. Protection is afforded for breaks inside or outside the containment even when it is assumed that there is a single failure in the steam line isolation system.

Feedwater Line Isolation

The feedwater lines are isolated upon actuation of the Safety Injection System in order to prevent excessive cooldown of the reactor coolant system. This mitigates the effects of an accident such as steam break which in itself causes excessive coolant temperature cooldown.

Feedwater line isolation also reduces the consequences of a steam line break inside the containment, by stopping the entry of feedwater.

Auxiliary Feedwater System Actuation

The automatic initiation of auxiliary feedwater flow to the steam generators by instruments identified in Table 3.7-2 ensures that the Reactor Coolant System Decay Heat can be removed following loss of main feedwater flow. This is consistent with the requirements of the "TMI-2 Lesson Learned Task Force Status Report", NUREG-0578, item 2.1.7.b.

Setting Limits

1. The high containment pressure limit is set at about 10% of design containment pressure. Initiation of Safety Injection protects against loss of coolant [2] or steam line break [3] accidents as discussed in the safety analysis.

2. The high-high containment pressure limit is set at about 50% of design containment pressure. Initiation of Containment Spray and Steam Line Isolation protects against large loss of coolant [2] or steam line break accidents [3] as discussed in the safety analysis.

3. The pressurizer low pressure setpoint for safety injection acutation is set substantially below system operating pressure limits. However, it is sufficiently high to protect against a loss-of-coolant accident as shown in the safety analysis. [2]

4. The steam line high differential pressure limit is set well below the differential pressure expected in the event of a large steam line break accident as shown in the safety analysis.[3]

5. The high steam line flow differential pressure setpoint is constant at 40% full flow between no load and 20% load and increasing linearly to 110% of full flow at full load in order to protect against large steam line break accidents. The coincident low T_{avg} setting limit for SIS and steam line isolation initiation is set below its hot shutdown value. The coincident steam line pressure setting limit is set below the full load operating pressure. The safety analysis shows that these settings provide protection in the event of a large steam line break.[3]

Automatic Function Operated from Radiation Monitors

The Process Radiation Monitoring System continuously monitors selected lines containing or possibly containing, radioactive effluent. Certain channels in this system actuate control valves on a high-activity alarm signal. Additional information on the Process Radiation Monitoring System is available in the FSAR.[4]

Accident Monitoring Instrumentation

The operability of the accident monitoring instrumentation is Table 3.7-6 ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. On the pressurizer PORV's, the pertinent channels consist of limit switch indication and acoustic monitor indication. The pressurizer safety valves utilize an acoustic monitor channel and a downstream high temperature indication channel. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975, and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations".

References

(1) FSAR - Section 7.5

(2) FSAR - Section 14.5

(3) FSAR - Section 14.3.2

(4) FSAR - Section 11.3.3

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
1.	Manual	1	--		Maintain hot shutdown
2.	Nuclear Flux Power Range	3	2	Low trip setting when 2 of 4 power channels greater than 10% of full power	Maintain hot shutdown
3.	Nuclear Flux Intermediate Range	1	--	2 of 4 power channels greater than 10% full power	Maintain hot shutdown
4.	Nuclear Flux Source Range	1	--	1 of 2 intermediate range channels greater than 10^{-10} amps	Maintain hot shutdown
5.	Overtemperature ΔT	2	1		Maintain hot shutdown
6.	Overpower ΔT	2	1		Maintain hot shutdown
7.	Low Pressurizer Pressure	2	1	3 of 4 nuclear power channels and 2 of 2 turbine load channels less than 10% of rated power	Maintain hot shutdown
8.	Hi Pressurizer Pressure	2	1	Same as Item 7 above	Maintain hot shutdown

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
9.	Pressurizer-Hi Water Level	2	1	3 of 4 nuclear power channels and 2 of 2 turbine load channels less than 10% of rated power	Maintain hot shutdown
10.	Low Flow	2/operable loop		If inoperable loop channels are not in service they must be placed in the tripped mode	Maintain hot shutdown
11.	Turbine Trip	2	1		Maintain less than 10% rated power
12.	Lo-Lo Steam Generator Water Level	2/non-isolated loop	1/non-isolated loop		Maintain hot shutdown
13.	Underfrequency 4KV Bus	2	1		Maintain hot shutdown
14.	Undervoltage 4KV Bus	2	1		Maintain hot shutdown

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	1	2	3	4
FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
15. Control rod misalignment Monitor**				
a) rod position deviation	1	-		Log individual rod positions once/hour, and after a load change > 10% or after > 30 inches of control rod motion.
b) quadrant power tilt monitor (upper and lower excore neutron detectors)	1	-		Log individual upper upper and lower ion chamber currents once/hour and after a load change > 10% or after > 30 inches of control rod motion.
16. Safety Injection	See Item 1 of TS Table 3.7-2			

TABLE 3.7-1
REACTOR TRIP

INSTRUMENT OPERATING CONDITIONS

	1	2	3	4
FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUN-DANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
17. Low steam generator water level with steam/feedwater mismatch flow	1/non-iso-lated loop 1/non-iso-lated loop	-- --		Maintain hot shutdown

**If both rod misalignment monitors (a and b) inoperable for 2 hours or more, the nuclear overpower trip shall be reset to 93 percent of rated power in addition to the increased surveillance noted.

TABLE 3.7-2
ENGINEERED SAFEGUARDS ACTION

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUN-DANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
	1. SAFETY INJECTION				
a.	Manual	1	0		Cold shutdown
b.	High Containment Press.	3	1		Cold shutdown
c.	High Differential Press. between any Steam Line and the Steam Line Header	2/non-isolated loop	1/non-isolated loop	Primary Pressure less than 2000 psig except when reactor is critical	Cold shutdown
d.	Pressurizer Low-Low Press.	2	1	Primary Pressure less than 2000 psig except when reactor is critical	Cold shutdown
e.	High Steam Flow in 2/3 Steam Lines with Low T_{avg} or Low Steam Line Press.	1/steamline 2 T_{avg} signals 2 Steam Press. Signals	*** 1 1	Reactor Coolant average temperature less than 543°F (nominal) during heatup and cooldown	Cold shutdown

***With the specified minimum operable channels the 2/3 high steam flow is already in the trip mode.

TABLE 3.7-2

EI[illegible]ERED SAFEGUARDS ACTION

INSTRUMENT OPERATING CONDITIONS

		1	2	3	4
	FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUNDANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
	2. CONTAINMENT SPRAY				
a.	Manual	2	**		Cold shutdown
b.	High Containment Press. (Hi-Hi Setpoint)	3	1		Cold shutdown
	3. AUXILIARY FEEDWATER				
a.	Steam Generator Water Level Low-Low				
	i. Start Motor Driven Pumps	2/Stm. Gen.	1	Loop Stop Valve in respective loop closed	Place inoperable channel in Tripped condition within one hour
	II. Start Turbine Driven Pumps	2/Stm. Gen.	1		
b.	RCP Undervoltage Start Turbine Driven Pump	2	1		Place inoperable channel in Tripped condition within one hour
c.	Safety Injection Start Motor Driven Pumps	(All safety injection initiating functions and requirements)			

**Must actuate 2 switches simultaneously.

TABLE 3.7-2

ENGINEERED SAFEGUARDS ACTION

INSTRUMENT OPERATING CONDITONS

		1	2	3	4
	FUNCTIONAL UNIT	MIN. OPERABLE CHANNELS	DEGREE OF REDUN-DANCY	PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDI-TIONED BY COLUMN 3 CANNOT BE MET
d.	Station Blackout Start Motor Driven Pump	2	0		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours and in cold shutdown within the following 30 hours.
e.	Trip of Main Feedwater Pumps Start Motor Pumps	1/Pump	1/Pump		Restore inoperable channel within 48 hours or be in hot shutdown within next 6 hours and in cold shutdown within the following 30 hours.

TABLE 3.7-3

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

INSTRUMENT OPERATING CONDITIONS

	FUNCTIONAL UNIT	1 MIN. OPERABLE CHANNELS	2 DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 EXCEPT AS CONDITIONED BY COLUMN 3 CANNOT BE MET
1.	CONTAINMENT ISOLATION				
a.	Safety Injection	See Item No. 1 of Table 3.7-2			Cold shutdown
b.	Manual	1	--		Hot shutdown
c.	High Containment Press. (Hi Setpoint)	3	1		Cold shutdown
d.	High Containment Press.	3	1		Cold shutdown
2.	STEAM LINE ISOLATION				
a.	High Steam Flow in 2/3 lines and 2/3 Low T_{avg} or 2/3 Low Steam Pressure	1/steamline 2/T_{avg} signals 2 Stm. Press. signals	*** 1 1		Cold shutdown
b.	High Containment Press. (Hi-Hi Level)	3	1		Cold shutdown
c.	Manual	1/line	--		Hot shutdown
3.	FEEDWATER LINE ISOLATION				
a.	Safety Injection	See Item No. 1 of Table 3.7-2			Cold shutdown

***With the specified minimum operable channels the 2/3 high steam flow is already in the trip mode

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
1	High Containment Pressure (High Containment Pressure Signal)	a) Safety Injection b) Containment Vacuum Pump Trip c) High Press. Containment Iso. d) Safety Injection Contain. Iso. e) F.W. Line Isolation	≤5 psig
2	High High Containment Pressure (High High Containment Pressure Signals)	a) Containment Spray b) Recirculation Spray c) Steam Line Isolation d) High High Press. Contain. Iso.	≤25 psig
3	Pressurizer Low Low Pressure	a) Safety Injection b) Safety Injection Cont. Iso. c) Feedwater Line Isolation	≥1,700 psig
4	High Differential Pressure Between Steam Line and the Steam Line Header	a) Safety Injection b) Safety Injection Contain. Iso. c) F.W. Line Isolation	≤150 psi
5	High Steam Flow in 2/3 Steam Lines	a) Safety Injection	≤40% (at zero load) of full steam flow ≤40% (at 20% load) of full steam flow
		b) Steam Line Isolation c) Safety Injection Contain. Iso. d) F.W. Line Isolation	≤110% (at full load) of full steam flow
	Coincident with Low T_{avg} or Low Steam		≥541°F T_{avg}
	Line Pressure		≥500 psig steam line pressure

TABLE 3.7-4

ENGINEERED SAFETY FEATURE SYSTEM INITIATION LIMITS INSTRUMENT SETTING

No.	FUNCTIONAL UNIT	CHANNEL ACTION	SETTING LIMIT
6	AUXILIARY FEEDWATER		
a.	Steam Generator Water Level Low-Low	Aux. Feedwater Initiation S/G Blowdown Isolation	≥5% narrow range
b.	RCP Undervoltage	Aux. Feedwater Initiation	≥70% nominal
c.	Safety Injection	Aux. Feedwater Initiation	All S.I. setpoints
d.	Station Blackout	Aux. Feedwater Initiation	≥46.7% nominal
e.	Main Feedwater Pump Trip	Aux. Feedwater Initiation	N.A.

TABLE 3.7-5

AUTOMATIC FUNCTIONS
OPERATED FROM RADIATION MONITORS ALARM

	MONITOR CHANNEL	AUTOMATIC FUNCTION AT ALARM CONDITIONS	MONITORING REQUIREMENTS	ALARM SETPOINT µCI/cc
1.	Process vent particulate and gas monitors (RM-GW-101 & RM-GW-102)	Stops discharge from contain. vacuum systems and waste gas decay tanks (shuts Valve Nos. RCV-GW-160, FCV-GW-260, FCV-GW-101)	See Specifications 3.11 and 4.9	Particulate $\leq 4\mathrm{x}10^{-}$ Gas $\leq 9\mathrm{x}10^{-2}$
2.	Component cooling water radiation monitors (RM-CC-105 & RM-CC-106)	Shuts surge tank vent valve HCV-CC-100	See Specifications 3.13 and 4.9	≤Twice Background
3.	Liquid waste disposal radiation monitors (RM-LW-108)	Shuts effluent discharge valves FCV-LW-104A and FCV-LW-104B	See Specifications 3.11 and 4.9	$\leq 1.5\mathrm{x}10^{-3}$
4.	Condenser air ejector radiation monitors (RM-SV-111 & RM-SV-211)	Diverts flow to the containment of the affected unit (Opens TV-SV-102 and shuts TV-SV-103 or opens TV-SV-202 and shuts TV-SV-203)	See Specification 3.11 and 4.9	≤1.3
5.	Containment particulte and gas monitors (RM-RMS-159 & RM-RMS-160, RM-RMS-259 & RM-RMS-260)	Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D)	See Specifications 3.10 and 4.0	Particulate $\leq 9\mathrm{x}10^{-}$ Gas $\leq 1\mathrm{x}10^{-5}$
6.	Manipulator crane area monitors (RM-RMS-162 & RM-RMS-262)	Trips affected unit's purge supply and exhaust fans, closes affected unit's purge air butterfly valves (MOV-VS-100A, B, C & D or MOV-VS-200A, B, C & D	See Specifications 3.10 and 4.9	≤50 mrem/hr

TABLE 3.7-6

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Auxiliary Feedwater Flow Rate	1 per S/G	1 per S/G
2.	Reactor Coolant System Subcooling Margin Monitor	2	1
3.	PORV Position Indicator (Primary Detector)	1/valve	1/valve
4.	PORV Position Indicator (Backup Detector)	1/valve	0
5.	PORV Block Valve Position Indicator	1/valve	1/valve
6.	Safety Valve Position Indicator (Primary Detector)	1/valve	1/valve
7.	Safety Valve Position Indicator (Backup Detector)	1/valve	0

3.8 CONTAINMENT

Applicability

Applies to the integrity and operating pressure of the reactor containment.

Objective

To define the limiting operating status of the reactor containment for unit operation.

Specification

A. Containment Integrity and Operating Pressure

1. The containment integrity, as defined in TS Section 1.0, shall not be violated, except as specified in Specification 3.8.A.2, below, unless the reactor is in the cold shutdown condition.
2. The reactor containment shall not be purged while the reactor is operating, except as stated in Specification 3.8.A.3.
3. During the plant startup, the remote manual valve on the steam jet air ejector suction line may be open, if under administrative control, while containment vacuum is being established. The Reactor Coolant System temperature and pressure must not exceed 350°F and 450 psig, respectively, until the air partial pressure in the containment has been reduced to a value equal to, or below, that specified in TS Fig. 3.8-1.
4. The containment integrity shall not be violated when the reactor vessel head is unbolted unless a shutdown margin greater than 10 percent $\Delta k/k$ is maintained.

5. Positive reactivity changes shall not be made by rod drive motion or boron dilution unless the containment integrity is intact.

6. The containment isolation valves shall be listed in Tables 3.8-1 and 3.8-2.

B. Internal Pressure

1. If the internal air partial pressure rises to a point 0.25 psi above the allowable value of the air partial pressure (TS Fig. 3.8-1), the reactor shall be brought to the hot shutdown condition.

2. If the leakage condition cannot be corrected without violating the containment integrity or if the internal partial pressure continues to rise, the reactor shall be brought to the cold shutdown condition utilizing normal operating procedures.

3. If the internal pressure falls below 8.25 psia the reactor shall be placed in the cold shutdown condition.

4. If the air partial pressure cannot be maintained greater than or equal to 9.0 psia, the reactor shall be brought to the hot shutdown condition.

Basis

The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure build-up in the containment if there is a loss-of-coolant accident.

The shutdown margins are selected based on the type of activities that are being carried out. The 10 percent Δk/k shutdown margin during refueling precludes criticality under any circumstance, even though fuel and control rod assemblies are being moved.

The allowable value for the containment air partial pressure is presented in TS Fig. 3.8-1 for service water temperatures from 25 to 90°F. The allowable value varies as shown in TS Fig. 3.8-1 for a given containment average temperature. The RWST water shall have a maximum temperature of 45°F.

The horizontal limit lines in TS Fig. 3.8-1 are based on LOCA peak calculated pressure criteria, and the sloped line is based on LOCA subatmospheric peak pressure criteria.

The curve shall be interpreted as follows:

> The horizontal limit line designates the allowable air partial pressure value for the given average containment temperature. The horizontal limit line applies for service water temperatures from 25°F to the sloped line intersection value (maximum service water temperature).

From TS Fig. 3.8-1, if the containment average temperature is 112°F and the service water temperature is less than or equal to 83°F, the allowable air partial pressure value shall be less than or equal to 9.65 psia. If the average containment temperature is 116°F and the service water temperature is less than or equal to 88°F, the allowable air partial pressure value shall be less than or equal to 9.35 psia. These horizontal limit lines are a result of the higher allowable initial containment average temperatures and the analysis of the pump suction break.

If the containment air partial pressure rises to a point 0.25 psi 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 0.25 psi above the allowable value, the maximum containment pressure will be less than 45 psig, the containment will depressurize in less than 1 hour, and the maximum subatmospheric peak pressure will be less than 0.0 psig.

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

FSAR Section 4.3.2	Reactor Coolant Pump
FSAR Section 5.2	Containment Isolation
FSAR Section 5.2.1	Design Bases
FSAR Section 5.5.2	Isolation Design

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION
A.	PHASE I CONTAINMENT ISOLATION (SAFETY INJECTION SIGNAL)	
1.	MOV-1867C	Boron Injection Tank Outlet
2.	MOV-1867D	Boron Injection Tank Outlet
3.	MOV-1289A	Charging Line
4.	MOV-1381	Reactor Coolant Pump Seal Water Return
5.	HCV-1200A	Letdown Orifice Isolation
6.	HCV-1200B	Letdown Orifice Isolation
7.	HCV-1200C	Letdown Orifice Isolation
8.	TV-SI-101A	Accumulator N_2 Relief Line
9.	TV-SI-101B	Accumulator N_2 Relief Line
10.	TV-SI-100	Accumulator N_2 Relief Line
11.	TV-VG-109A	Primary Drain Transfer Tank Vent
12.	TV-VG-109B	Primary Drain Transfer Tank Vent
13.	TV-DG-108A	Primary Drain Transfer Pump Discharge
14.	TV-DG-108B	Primary Drain Transfer Pump Discharge
15.	TV-CC-109A*	Component Cooling from RHR's
16.	TV-CC-109B*	Component Cooling from RHR's
17.	TV-SS-100A	Pressurizer Liquid Sample
18.	TV-SS-100B	Pressurizer Liquid Sample
19.	TV-SS-101A	Pressurizer Vapor Sample
20.	TV-SS-101B	Pressurizer Vapor Sample

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
21.	TV-SS-103	Residual Heat Removal System Sample
22.	TV-SS-106A	Reactor Coolant Hot Leg Sample
23.	TV-SS-106B	Reactor Coolant Hot Leg Sample
24.	TV-SS-102A	Reactor Coolant Cold Leg Sample
25.	TV-SS-102B	Reactor Coolant Cold Leg Sample
26.	TV-SS-104A	Pressurizer Relief Tank Vapor Sample
27.	TV-SS-104B	Pressurizer Relief Tank Vapor Sample
28.	TV-CH-1204	Letdown Isolation Valve
29.	TV-PG-1519A	Primary Grade Water to Pressurizer Relief Tank
30.	TV-BD-100A*	Steam Generator Blowdown Valve
31.	TV-BD-100B*	Steam Generator Blowdown Valve
32.	TV-BD-100C*	Steam Generator Blowdown Valve
33.	TV-BD-100D*	Steam Generator Blowdown Valve
34.	TV-BD-100E*	Steam Generator Blowdown Valve
35.	TV-BD-100F*	Steam Generator Blowdown Valve
36.	TV-DA-100A	Containment Sump Pump Isolation
37.	TV-DA-100B	Containment Sump Pump Isolation
38.	TV-MS-109*	Main Steam Drain Trip Valve
39.	TV-MS-110*	Main Steam Drain Trip Valve
40.	TV-LM-100A	Containment Isolation Monitoring
41.	TV-LM-100B	Containment Isolation Monitoring
42.	TV-LM-100C	Containment Isolation Monitoring

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
43.	TV-LM-100D	Containment Isolation Monitoring
44.	TV-LM-100E	Containment Isolation Monitoring
45.	TV-LM-100F	Containment Isolation Monitoring
46.	TV-LM-100G	Containment Isolation Monitoring
47.	TV-LM-100H	Containment Isolation Monitoring
48.	TV-CV-150A	Containment Vacuum Suction Valve
49.	TV-CV-150B	Containment Vacuum Suction Valve
50.	TV-LM-101A	Leakage Monitoring Sealed Reference
51.	TV-LM-101B	Leakage Monitoring Sealed Reference
52.	TV-CV-150C	Containment Vacuum Suction Valve
53.	TV-CV-150D	Containment Vacuum Suction Valve
54.	TV-SV-102A	Condenser Air Ejector Vent Trip Valve

B. PHASE II CONTAINMENT ISOLATION (HI CLS SIGNAL)

1.	TV-RM-100A	Containment Air & Particulate Rad. Mon. TV's
2.	TV-RM-100B	Containment Air & Particulate Rad. Mon. TV's
3.	TV-RM-100C	Containment Air & Particulate Rad. Mon. TV's
4.	TV-IA-101A	Containment Instr. Air Compressor Suction
5.	TV-IA-101B	Containment Instr. Air Compressor Suction

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
C.	PHASE III CONTAINMENT ISOLATION (HI-HI CLS SIGNAL)	
1.	TV-MS-101A*	Main Steam Trip Valve
2.	TV-MS-101B*	Main Steam Trip Valve
3.	TV-IA-100	Containment Instr. Air Compressor Disch. Vlv.
4.	TV-MS-101C*	Main Steam Trip Valve
5.	TV-CC-107*	CC from RCP Thermal Barriers
6.	TV-CC-101A*	CC from A Air Recirc.
7.	TV-CC-101B*	CC from B Air Recirc.
8.	TV-CC-101C*	CC from C Air Recirc.
9.	TV-CC-105A*	CC from "A" RCP
10.	TV-CC-105B*	CC from "B" RCP
11.	TV-CC-105C*	CC from "C" RCP
D.	CONTAINMENT PURGE & EXHAUST	
1.	MOV-VS-100C	R.C. Purge Exhaust MOV's
2.	MOV-VS-100D	R.C. Purge Exhaust MOV's
3.	MOV-VS-101	R.C. Purge Exhaust Bypass MOV
4.	MOV-VS-100A	R.C. Purge Supply MOV's
5.	MOV-VS-100B	R.C. Purge Supply MOV's
6.	MOV-VS-102	Contain. Vacuum Breaker Atmos. Supply MOV

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

	VALVE NUMBER	FUNCTION
E.	REMOTE MANUAL VALVES	
1.	MOV-CS-101A	Containment Spray Discharge Valve
2.	MOV-CS-101B	Containment Spray Discharge Valve
3.	MOV-CS-101C	Containment Spray Discharge Valve
4.	MOV-CS-101D	Containment Spray Discharge Valve
5.	MOV-RS-155A	Outside Recirc. Spray Suction Valve
6.	MOV-RS-155B	Outside Recirc. Spray Suction Valve
7.	MOV-RS-156A	Outside Recirc. Discharge Valve
8.	MOV-RS-156B	Outside Recirc. Discharge Valve
9.	MOV-1842	Bypasses Boron Injec. Tank to Cold Leg Injec.
10.	MOV-RH-100	Resi. Heat Remov. to RWST
11.	FCV-1160	Loop Fill Header Flow Valve
12.	MOV-1890A	Lo Header S. I. Pump Disch. from Hot Leg
13.	MOV-1890B	Lo Header S. I. Pump Disch. from Hot Leg
14.	MOV-1890C	Lo Header S. I. Pump Disch. from Cold Leg
15.	MOV-1869A	Iso. from Hot Leg to Hi Header S. I. Line A
16.	MOV-1869B	Iso. from Hot Leg to Hi Header S. I. Line B
17.	MOV-1860A	Iso. from Sump to Lo Header S. I.
18.	MOV-1860B	Iso. Valve from Sump to Lo Header S. I.
19.	MOV-SW-104A*	SW to "A" HX's
20.	MOV-SW-104B*	SW to "B" HX's
21.	MOV-SW-104C*	SW to "C" HX's

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER	FUNCTION
22. MOV-SW-104D*	SW to "D" HX's
23. MOV-SW-105A*	SW from "A" HX's
24. MOV-SW-105B*	SW from "B" HX's
25. MOV-SW-105C*	SW from "C" HX's
26. MOV-SW-105D*	SW from "D" HX's
27. HCV-CV-100	Cont. Vacuum Isolation

F. MANUAL VALVES

1. 1-SI-150	Boron Injection Tank 1" line
2. 1-SI-32	Accumulator Fill Valve
3. 1-GW-182	Discharge from Hydrogen Analyzer
4. 1-GW-183	Discharge from Hydrogen Analyzer
5. 1-SA-60	Service Air to Containment
6. 1-SA-62	Service Air to Containment
7. 1-IA-446	Instrument Air to Containment
8. 1-VA-1	Outside Isolation from Primary Vent Pot
9. 1-VA-6	Inside Isolation from Primary Vent Pot
10. 2-IA-446	Cross Tie from #2 Instrument Air Header
11. 1-GW-175	Suction from Containment to H_2 Analyzer
12. 1-GW-166	Suction from Containment to H_2 Analyzer
13. 1-GW-174	Inlet to Cont. from H_2 Analyzer Outside Cont.
14. 1-FP-151	Outside Iso. Vlv for Cont. Fire Protection
15. 1-FP-152	Outside Iso. Vlv for Cont. Fire Protection

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
16.	1-RL-3	Inlet Vlv to Cavity from RCS Outside Cont.
17.	1-RL-5	Inlet Vlv to Cavity from RCS Inside Cont.
18.	1-RL-13	Suction Vlv to 1-RL-P-1A Inside Containment
19.	1-RL-15	Suction Vlv to 1-RL-P-1A Outside Containment
20.	1-SI-73	Accumulator N_2 Fill Vlv Outside Containment
21.	1-SI-174	Bypasses MOV-1869A
22.	1-SW-208	RS HX SW Drain
23.	1-SW-106	RS HX SW Drain
24.	1-CV-2	Cont. Vacuum Isolation

G. CONTAINMENT CHECK VALVES

1.	1-FP-153	Inside Cont. - Fire Protection Header
2.	1-VP-12	Inside Cont. - Air Eject Disch to Cont.
3.	1-RS-17	Inside Cont. - RS Disch to Cont. A
4.	1-RS-11	Inside Cont. - RS Disch to Cont. B
5.	1-CS-13	Inside Cont. - Discharge of 1-CS-P-1A
6.	1-CS-24	Inside Cont. - Discharge of 1-CS-P-1B
7.	1-IA-938	Inside Cont. - Disch of Cont. IA Component
8.	2-IA-446	Manual Valve - Disch. of IA Component Unit #2
9.	1-SI-234	Check Inside Cont. - N_2 to Accumulator
10.	1-IA-939	Check Inside Cont. - Disch. of Cont. IA Component Unit #1
11.	1-IA-446	Manual Vlv - Disch. of Unit 1 Instr. Air Comp.

TABLE 3.8-1**

UNIT NO. 1 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
12.	1-RC-160	Check Valve Inside Contain. from PG Supply
13.	1-RM-3	Check Valve Inside Contain. - Rad. Monitoring Suc.
14.	1-IA-939	Instr. Air Check Valve to Containment
15.	1-SA-446	Service Air Check Valve to Containment
16.	1-CC-177*	CC to "A" RHR HX
17.	1-CC-176*	CC to "B" RHR HX
18.	1-SI-225	HHSI from BIT
19.	1-CC-242*	CC to "A" Air Recirc.
20.	1-CC-233*	CC to "B" Air Recirc.
21.	1-CC-224*	CC to "C" Air Recirc.
22.	1-CH-309	Normal Chg. Hdr
23.	1-CC-1*	CC to "A" RCP
24.	1-CC-58*	CC to "B" RCP
25.	1-CC-59*	CC to "C" RCP
26.	1-SI-224	HHSI BIT Bypass
27.	1-SI-226	HHSI to Hot Legs
28.	1-SI-228	LHSI Pp Discharge
29.	1-SI-229	LHSI Pp Discharge
30.	1-SI-227	LHSI to Hot Leg

* - Not subject to Type "C" Testing.

** - Modifications to this table should be submitted to the NRC as part of the next license amendment.

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION
A.	PHASE I CONTAINMENT ISOLATION (SAFETY INJECTION SIGNAL)	
1.	MOV-2867C	Boron Injection Tank Outlet
2.	MOV-2867D	Boron Injection Tank Outlet
3.	MOV-2289A	Charging Line
4.	MOV-2381	Reactor Coolant Pump Seal Water Return
5.	HCV-2200A	Letdown Orifice Isolation
6.	HCV-2200B	Letdown Orifice Isolation
7.	HCV-2200C	Letdown Orifice Isolation
8.	TV-SI-201A	Accumulator N_2 Relief Line
9.	TV-SI-201B	Accumulator N_2 Relief Line
10.	TV-SI-200	Accumulator N_2 Relief Line
11.	TV-VG-209A	Primary Drain Transfer Tank Vent
12.	TV-VG-209B	Primary Drain Transfer Tank Vent
13.	TV-DG-208A	Primary Drain Transfer Pump Discharge
14.	TV-DG-208B	Primary Drain Transfer Pump Discharge
15.	TV-CC-209A*	Component Cooling from RHR's
16.	TV-CC-209B*	Component Cooling from RHR's
17.	TV-SS-200A	Pressurizer Liquid Sample
18.	TV-SS-200B	Pressurizer Liquid Sample
19.	TV-SS-201A	Pressurizer Vapor Sample
20.	TV-SS-201B	Pressurizer Vapor Sample

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

	VALVE NUMBER	FUNCTION
21.	TV-SS-203	Residual Heat Removal System Sample
22.	TV-SS-206A	Reactor Coolant Hot Leg Sample
23.	TV-SS-206B	Reactor Coolant Hot Leg Sample
24.	TV-SS-202A	Reactor Coolant Cold Leg Sample
25.	TV-SS-202B	Reactor Coolant Cold Leg Sample
26.	TV-SS-204A	Pressurizer Relief Tank Vapor Sample
27.	TV-SS-204B	Pressurizer Relief Tank Vapor Sample
28.	TV-CH-2204	Letdown Isolation Valve
29.	TV-PG-2519A	Primary Grade Water to Pressurizer Relief Tank
30.	TV-BD-200A*	Steam Generator Blowdown Valve
31.	TV-BD-200B*	Steam Generator Blowdown Valve
32.	TV-BD-200C*	Steam Generator Blowdown Valve
33.	TV-BD-200D*	Steam Generator Blowdown Valve
34.	TV-BD-200E*	Steam Generator Blowdown Valve
35.	TV-BD-200F*	Steam Generator Blowdown Valve
36.	TV-DA-200A	Containment Sump Pump Isolation
37.	TV-DA-200B	Containment Sump Pump Isolation
38.	TV-MS-209*	Main Steam Drain Trip Valve
39.	TV-MS-210*	Main Steam Drain Trip Valve
40.	TV-LM-200A	Containment Isolation Monitoring
41.	TV-LM-200B	Containment Isolation Monitoring
42.	TV-LM-200C	Containment Isolation Monitoring

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
43.	TV-LM-200D	Containment Isolation Monitoring
44.	TV-LM-200E	Containment Isolation Monitoring
45.	TV-LM-200F	Containment Isolation Monitoring
46.	TV-LM-200G	Containment Isolation Monitoring
47.	TV-LM-200H	Containment Isolation Monitoring
48.	TV-CV-250A	Containment Vacuum Suction Valve
49.	TV-CV-250B	Containment Vacuum Suction Valve
50.	TV-LM-201A	Leakage Monitoring Sealed Reference
51.	TV-LM-201B	Leakage Monitoring Sealed Reference
52.	TV-CV-250C	Containment Vacuum Suction Valve
53.	TV-CV-250D	Containment Vacuum Suction Valve
54.	TV-SV-202A	Condenser Air Ejector Vent Trip Valve

B. PHASE II CONTAINMENT ISOLATION
(HI CLS SIGNAL)

1.	TV-RM-200A	Containment Air & Particulate Rad. Mon. TV's
2.	TV-RM-200B	Containment Air & Particulate Rad. Mon. TV's
3.	TV-RM-200C	Containment Air & Particulate Rad. Mon. TV's
4.	TV-IA-201A	Containment Instr. Air Compressor Suction
5.	TV-IA-201B	Containment Instr. Air Compressor Suction

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

	VALVE NUMBER	FUNCTION
C.	PHASE III CONTAINMENT ISOLATION (HI-HI CLS SIGNAL)	
1.	TV-MS-201A*	Main Steam Trip Valve
2.	TV-MS-201B*	Main Steam Trip Valve
3.	TV-IA-200	Containment Instr. Air Compressor Disch. Vlv.
4.	TV-MS-201C*	Main Steam Trip Valve
5.	TV-CC-207*	CC from RCP Thermal Barriers
6.	TV-CC-201A*	CC from A Air Recirc.
7.	TV-CC-201B*	CC from B Air Recirc.
8.	TV-CC-201C*	CC from C Air Recirc.
9.	TV-CC-205A*	CC from "A" RCP
10.	TV-CC-205B*	CC from "B" RCP
11.	TV-CC-205C*	CC from "C" RCP
D.	CONTAINMENT PURGE & EXHAUST	
1.	MOV-VS-200C	R.C. Purge Exhaust MOV's
2.	MOV-VS-200D	R.C. Purge Exhaust MOV's
3.	MOV-VS-201	R.C. Purge Exhaust Bypass MOV
4.	MOV-VS-200A	R.C. Purge Supply MOV's
5.	MOV-VS-200B	R.C. Purge Supply MOV's
6.	MOV-VS-202	Contain. Vacuum Breaker Atmos. Supply MOV

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
E.	REMOTE MANUAL VALVES	
1.	MOV-CS-201A	Containment Spray Discharge Valve
2.	MOV-CS-201B	Containment Spray Discharge Valve
3.	MOV-CS-201C	Containment Spray Discharge Valve
4.	MOV-CS-201D	Containment Spray Discharge Valve
5.	MOV-RS-255A	Outside Recirculation Spray Suction Valve
6.	MOV-RS-255B	Outside Recirc. Spray Suction Valve
7.	MOV-RS-256A	Outside Recirc. Discharge Valve
8.	MOV-RS-256B	Outside Recirc. Discharge Valve
9.	MOV-2842	Bypasses Boron Injec. Tank to Cold Leg Injec.
10.	MOV-RH-200	Resi. Heat Remov. to RWST
11.	FCV-2160	Loop Fill Header Flow Valve
12.	MOV-2890A	Lo Header S.I. Pump Disch. from Hot Leg
13.	MOV-2890B	Lo Header S.I. Pump Disch. from Hot Leg
14.	MOV-2890C	Lo Header S.I. Pump Disch. from Cold Leg
15.	MOV-2869A	Iso. from Hot Leg to Hi Header S. I. Line A
16.	MOV-2869B	Iso. from Hot Leg to Hi Header S. I. Line B
17.	MOV-2860A	Iso. from Sump to Lo Header S. I.
18.	MOV-2860B	Iso. Valve from Sump to Lo Header S. I.
19.	MOV-SW-204A*	SW to "A" HX's
20.	MOV-SW-204B*	SW to "B" HX's
21.	MOV-SW-204C*	SW to "C" HX's

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
22.	MOV-SW-204D*	SW to "D" HX's
23.	MOV-SW-205A*	SW from "A" HX's
24.	MOV-SW-205B*	SW from "B" HX's
25.	MOV-SW-205C*	SW from "C" HX's
26.	MOV-SW-205D*	SW from "D" HX's
27.	HCV-CV-200	Cont. Vacuum Isolation

F. MANUAL VALVES

1.	2-SI-150	Boron Injection Tank 1" line
2.	2-SI-32	Accumulator Fill Valve
3.	2-GW-182	Discharge from Hydrogen Analyzer
4.	2-GW-183	Discharge from Hydrogen Analyzer
5.	2-SA-60	Service Air
6.	2-SA-62	Service Air
7.	2-IA-446	Instrument Air to Containment
8.	2-VA-1	Outside Isolation from Primary Vent Pot
9.	2-VA-6	Inside Isolation from Primary Vent Pot
10.	2-IA-446	Cross Tie from #1 Instrument Air Header
11.	2-GW-175	Suction from Cont. to H_2 Analyzer
12.	2-GW-166	Suction from Cont. to H_2 Analyzer
13.	2-GW-174	Inlet to Cont. from H_2 Analyzer Outside Cont.
14.	2-FP-151	Outside Iso. Vlv for Cont. Fire Protection
15.	2-FP-152	Outside Iso. Vlv for Cont. Fire Protection

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
16.	2-RL-3	Inlet Vlv to Cavity from RCS Outside Cont.
17.	2-RL-5	Inlet Vlv to Cavity from RCS Inside Cont.
18.	2-RL-13	Suction Vlv to 2-RL-P-1A Inside Containment
19.	2-RL-15	Suction Vlv to 2-RL-P-1A Outside Containment
20.	2-SI-73	Accumulator N_2 Fill Vlv Outside Containment
21.	2-SI-174	Bypasses MOV-1869A
22.	2-SW-208	RS HX SW Drain
23.	2-SW-106	RS HX SW Drain
24.	2-CV-2	Cont. Vacuum Isolation

G. CONTAINMENT CHECK VALVES

1.	2-FP-153	Inside Cont. - Fire Protection Header
2.	2-VP-12	Inside Cont. - Air Eject Disch to Cont.
3.	2-RS-17	Inside Cont. - RS Disch to Cont. A
4.	2-RS-11	Inside Cont. - RS Disch to Cont. B
5.	2-CS-13	Inside Cont. - Discharge of 2-CS-P-1A
6.	2-CS-24	Inside Cont. - Discharge of 2-CS-P-1B
7.	2-IA-938	Inside Cont. - Disch of Cont. IA Component
8.	2-IA-446	Manual Valve - Disch. of IA Component Unit #2
9.	2-SI-234	Check Inside Cont. - N_2 to Accumulator
10.	2-IA-939	Check Inside Cont. - Disch. of Cont. IA Component Unit #2
11.	2-IA-446	Manual Vlv - Disch. of Unit 2 Instr. Air Comp.

TABLE 3.8-2**

UNIT NO. 2 CONTAINMENT ISOLATION VALVES (Continued)

VALVE NUMBER		FUNCTION
12.	2-RC-160	Check Valve Inside Contain. from PG Supply
13.	2-RM-3	Check Valve Inside Contain. - Rad. Monitoring Suc.
14.	2-IA-939	Instr. Air Check Valve to Containment
15.	2-SA-446	Service Air Check Valve to Containment
16.	2-CC-177*	CC to "A" RHR HX
17.	2-CC-176*	CC to "B" RHR HX
18.	2-SI-225	HHSI from BIT
19.	2-CC-242*	CC to "A" Air Recirc.
20.	2-CC-233*	CC to "B" Air Recirc.
21.	2-CC-224*	CC to "C" Air Recirc.
22.	2-CH-309	Normal Chg. Hdr
23.	2-CC-1*	CC to "A" RCP
24.	2-CC-58*	CC to "B" RCP
25.	2-CC-59*	CC to "C" RCP
26.	2-SI-224	HHSI BIT Bypass
27.	2-SI-226	HHSI to Hot Legs
28.	2-SI-228	LHSI Pp Discharge
29.	2-SI-229	LHSI Pp Discharge
30.	2-SI-227	LHSI to Hot Leg

* - Not subject to Type "C" Testing.

** - Modifications to this table should be submitted to the NRC as part of the next license amendment.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

A. Calibration, testing, and checking of instrumentation channels shall be performed as detailed in Table 4.1-1.

B. Equipment tests shall be conducted as detailed below and in Table 4.1-2A.

1. Each Pressurizer PORV shall be demonstrated operable:

   a. At least once per 31 days by performance of a channel functional test, excluding valve operation, and

   b. At least once per 18 months by performance of a channel calibration.

2. Each Pressurizer PORV block valve shall be demonstrated operable at least once per 92 days by operating the valve through one complete cycle of full travel.

3. The pressurizer water volume shall be determined to be within its limit as defined in Specification 2.3.A.3.a at least once per 12 hours whenever the reactor is not subcritical by at least 1% $\Delta k/k$.

C. Sampling tests shall be conducted as detailed in Table 4.1-2B.

D. Whenever containment integrity is not required, only the asterisked items in Table 4.1-1 and 4.1-2A and 4.1-2B are applicable.

E. Flushing of sensitized stainless steel pipe sections shall be conducted as detailed in TS Table 4.1-3A and 4.1-3B.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECK, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
1.	Nuclear Power Range	S M(3)	D (1) Q (3)	BW(2)	1) Against a heat balance standard 2) Signal of ΔT; bistable action (permissive, rod stop, strips) 3) Upper and lower chambers for symetri offset by means of the moveable inco detector system.
2.	Nuclear Intermediate Range	*S(1)	N.A.	P(2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	*S(1)	N.A.	P(2)	1) Once/Shift when in service 2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	*S	R	BW(1) BW(2)	1) Overtemperature - ΔT 2) Overpower - ΔT
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure (High & Low)	S	R	M	
8.	4 Kv Voltage and Frequency	S	R	M	Reactor protection circuit only
9.	Analog Rod Position	*S(1,2) (4)	R	M(3)	1) With step counters 2) Each six inches of rod motion when data logger is out of service 3) Rod bottom bistable action 4) NA when reactor is in cold shut-down

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
10.	Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) Each six inches of rod motion when data logger is out of service 2) With analog rod position
11.	Steam Generator Level	S	R	M	
12.	Charging Flow	N.A.	R	N.A.	
13.	Residual Heat Removal Pump Flow	N.A	R	N.A.	
14.	Boric Acid Tank Level	*D	R	N.A.	
15.	Refueling Water Storage Tank Level	S	R	M	
16.	Boron Injection Tank Level	W	N.A.	N.A.	
17.	Volume Control Tank Level	N.A.	R	N.A.	
18.	Reactor Containment Pressure-CLS	*D	R	M(1)	1) Isolation Valve signal and spray signal
19.	Processing and Area Radiation Monitoring Systems	*D	R	M	
20.	Boric Acid Control	N.A.	R	N.A.	
21.	Containment Sump Level	N.A.	R	N.A.	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Containment Pressure-Vacuum Pump System	S	R	N.A.	
24.	Steam Line Pressure	S	R	M	

TABLE 4.1-1

	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	REMARKS
25.	Turbine First Stage Pressure	S	R	M	
26.	Emergency Plan Radiation Instr.	*M	R	M	
27.	Environmental Radiation Monitors	*M	N.A.	N.A.	TLD Dosimeters
28.	Logic Channel Testing	N.A.	N.A.	M	
29.	Turbine Overspeed Protection Trip Channel (Electrical)	N.A.	R	R	
30.	Turbine Trip Setpoint	N.A.	R	R	Stop valve closure or low EH fluid pressure
31.	Seismic Instrumentation	M	SA	M	
32.	Reactor Trip Breaker	N.A.	N.A.	M	
33.	Reactor Coolant Pressure (Low)	N.A.	R	N.A.	
34.	Auxiliary Feedwater				
	a. Steam Generator Water Level Low-Low	S	R	M	
	b. RCP Undervoltage	S	R	M	
	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	

S - Each shift M - Monthly
D - Daily P - Prior to each startup if not done previous week
W - Weekly R - Each Refueling Shutdown
NA - Not applicable BW - Every two weeks
SA - Semiannually AP - After each startup if not done previous week
Q - Every 90 effective full power days

* See Specification 4.1D

TABLE 4.1-2

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Auxiliary Feedwater Flow Rate	P	R
2.	Reactor Coolant System Subcooling Margin Monitor	M	R
3.	PORV Position Indicator (Primary Detector)	M	R
4.	PORV Position Indicator (Backup Detector)	M	R
5.	PORV Block Valve Position Indicator	M	R
6.	Safety Valve Position Indicator	M	R
7.	Safety Valve Position Indicator (Backup Detector)	M	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 and cold conditions	Each refueling shutdown or after disassembly or maintenance requiring the breech of the Reactor Coolant System integrity	7
2.	Control Room Assemblies	Partial movement of all rods	Every 2 weeks	7
3.	Refueling Water Chemical Addition Tank	Functional	Each refueling shutdown	6
4.	Pressurizer Safety Valves	Setpoint	Each refueling shutdown	4
5.	Main Steam Safety Valves	Setpoint	Each refueling shutdown	10
6.	Containment Isolation Trip	*Functional	Each refueling shutdown	5
7.	Refueling System Interlocks	*Functional	Prior to refueling	9.12
8.	Service Water System	*Functional	Each refueling shutdown	9.9
9.	Fire Protection Pump and Power Supply	Functional	Monthly	9.10
10.	Primary System Leakage	*Evaluate	Daily	4
11.	Diesel Fuel Supply	*Fuel Inventory	5 days/week	8.5
12.	Boric Acid Piping Heat Tracing Circuits	*Operational	Monthly	9.1
13.	Main Steam Line Trip	Functional (1) Full closure (2) Partial closure	 (1) Each cold shutdown (2) Before each startup	10

TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
14.	Service Water System Valves in Line Supplying Recirculation Spray Heat Exchangers	Functional	Each refueling	9.9
15.	Control Room Ventilation System	*Ability to maintain positive pressure for 1 hour using a volume of air equivalent to or less than stored in the bottled air supply	Each refueling interval (approx. every 12-18 months)	9.13
16.	Reactor Vessel Overpressure Mitigating System (except backup air supply)	Functional & Setpoint	Prior to decreasing RCS temperature below 350°F and monthly while the RCS is <350°F and the Reactor Vessel Head is bolted	None
17.	Reactor Vessel Overpressure Mitigating System Backup Air Supply	Setpoint	Refueling	None

TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
18. Primary Coolant System Pressure Isolation Valves	Functional	1. Periodic leakage testing [(a)(b)] on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

*See Specification 4.1.D.

H. Practice of site evacuation exercises shall be conducted annually, following emergency procedures and including a check of communications with off-site report groups. An annual review of the Emergency Plan will be performed.

I. The industrial security program which has been established for the station shall be implemented, and appropriate investigation and/or corrective action shall be taken if the provisions of the program are violated. An annual review of the program shall be performed.

J. The facility fire protection program and implementing procedures which have been established for the station shall be implemented. The program shall be reviewed at least once every two years.

K <u>Systems Integrity</u>

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and
2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

L. Iodine Monitoring

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.

(1) 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 totalling 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.

C. Monthly Operating Report: Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the Reactor Coolant System PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-280 AND 50-281

VIRGINIA ELECTRIC AND POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 72 to Facility Operating License No. DPR-32 and Amendment No. 73 to Facility Operating License No. DPR-37 issued to Virginia Electric and Power Company (the licensee), which revised Technical Specifications for operation of the Surry Power Station, Unit Nos. 1 and 2, respectively, (the facilities), located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments incorporate the requirements for implementation of the TMI-2 Lessons Learned Category "A" items. They specifically include the areas of emergency power supply requirements, valve position indication, instrumentation for inadequate core cooling, containment isolation, auxiliary feedwater systems, and the implementation of programs to reduce leakage outside containment and to accurately determine airborne iodine concentrations.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since these amendment do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 14, 1980, as supplemented December 23, 1980 and August 21, 1981, (2) Amendment Nos. 72 and 73 to License Nos. DPR-32 and DPR-37, (3) the Commission's related Safety Evaluation dated April 24, 1980 and (4) the Commission's letter dated September 29, 1981. All of these items are available for public inspection at the Commission's Public Document Room 1717 H Street, N. W., Washington, D. C. and at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185. Copies of items (2), (3) and (4) may be obtained upon request addressed to Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 29th day of September 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing