Vogtle
Electric Generating Plant
Unit 1 and Unit 2

Environmental Protection Plan

Georgia Power

the southern electric system

APPENDIX B

TO FACILITY OPERATING LICENSE NO._______

VOGTLE ELECTRIC GENERATING PLANT

UNIT 1 AND UNIT 2

GEORGIA POWER COMPANY

DOCKET NOS. 50-424, 50-425

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

VOGTLE ELECTRIC GENERATING PLANT

UNITS 1 AND 2

ENVIRONMENTAL PROTECTION PLAN

(NONRADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of nonradiological environmental values during operating of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement - Operating License Stage (FES-OL) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility operation and of actions taken to control those effects.

2.0 Environmental Protection Issues

In the FES-OL dated March, 1985, the staff considered the environmental impacts associated with the operation of the two unit Vogtle Electric Generating Plant (VEGP). Certain environmental issues were identified which required study or license conditions to resolve environmental concerns and to assure adequate protection of the environment.

2.1 Aquatic Issues

No specific aquatic issues were raised by the NRC staff in the FES-OL. Compliance with the NPDES permit will assure adequate protection of the aquatic environment.

2.2 Terrestrial Issues

No specific terrestrial issues were identified by the NRC staff in the FES-OL. Issues raised during the ASLB licensing hearings relative to cooling tower emmissions were fully resolved during that process and no further conditions are required.

2.3 Noise Issues

License conditions relative to noise associated with transmission facilities are specified in Section 4.2.4 of this EPP.

3.0 Consistency Requirements

3.1 Plant Design and Operation

The licensee may make changes in plant design or operation or perform tests or experiments affecting the environment provided such activities do not involve an unreviewed environmental question and do not involve a change in the EPP*. Changes in plant design or operation or performance of tests which do not affect the environment are not subject to the requirements of this EPP. Activities governed by Section 3.3 are not subject to the requirements of this Section.

A proposed change test or experiment shall be deemed to involve an unreviewed environmental question if it concerns: (1) a matter which may result in a significant increase in any adverse environmental impact previously evaluated in the FES-OL, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level; or (3) a matter, not previously reviewed and evaluate' in the documents specified in (1) of this Subsection, which may have a significant adverse environmental impact.

Before engaging in additional construction or operational activities which may significantly affect the environment, the licensee shall prepare and record an environmental evaluation of such activity. Activities are excluded from this requirement if all measureable nonradiological environmental effects are

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*This provision does not relieve the licensee of the requirements of 10 CFR 10.59.

confined to the on-site areas previously disturbed during site preparation and plant construction. When the evaluation indicates that such activity involves an unreviewed environmental question, the licensee shall provide a written evaluation of such activity and obtain prior NRC approval. When such activity involves a change in the EPP, such activity and change to the EPP may be implemented only in accordance with an appropriate license amendment as set forth in Section 5.3 of this EPP.

The licensee shall maintain records of changes in plant design or operation and of tests and experiments carried out pursuant to this Subsection. These records shall include written evaluations which provide bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of this EPP to meet the objectives specified in Section 1.0. The licensee shall include as part of this Annual Environmental Operating Report (per Subsection 5.4.1) brief descriptions, analyses, interpretations, and evaluations of such changes, tests, and experiments.

3.2 Reporting Related to the NPDES Permit and State Certification

Changes to, or renewals of, the NPDES Permits or the State certification shall be reported to the NRC within 30 days following the date the change or renewal is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The licensee shall notify the NRC of changes to the effective NPDES Permit proposed by the licensee by providing NRC with a copy of the proposed change immediately after it is submitted to the permitting agency. The licensee shall provide the NRC a copy of the application for renewal of the NPDES Permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in plant design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State, and local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to plant operation shall be recorded and reported to the NRC within 24 hours followed by a written report per Subsection 5.4.2. The following are examples: excessive bird impaction events, onsite plant or animal disease outbreaks, mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973, fish kills, increase in nuisance organisms or conditions, and unanticipated or emergency discharge of waste water or chemical substances.

No route monitoring programs are required to implement this condition.

4.2 Environmental Monitoring

4.2.1 Aquatic Monitoring

The certifications and permits required under the Clean Water Act provide mechanisms for protecting water quality and, indirectly, aquatic biota. The NRC will rely on the decisions made by the State of Georgia, under the authority of the Clean Water Act, for any requirements for aquatic monitoring.

4.2.2 Terrestrial Monitoring

No terrestrial monitoring is required.

4.2.3 Maintenance of Transmission Line Corridors

The use of herbicides within the Vogtle Electric Generating Plant transmission line corridors (VEGP-Thalmann, VEGP-Scherer, Georgia side of VEGP-South Carolina Electric and Gas, and VEGP-Goshen) shall conform to the approved use of selected herbicides as registered by the Environmental Protection Agency and approved by the State of Georgia authorities and applied as directed on the herbicide label.

Records shall be maintained in accordance with EPA or State of Georgia requirements by the licensee's Transmission Operating and Maintenance Department concerning herbicide use. Such records shall be made readily available to the NRC upon request. There shall be no routine reporting requirement associated with this condition.

4.2.3.1 Ebenezer Creek

Any routine maintenance involving trimming of the trees within the National Natural Landmark area necessary to maintain conductor clearance shall be done by hand (Section 5.2.2, FES-OL).

4.2.3.2 Francis Plantation

Routine maintenance involving trimming of the trees within the National Register of Historic Places property necessary to maintain conductor clearance shall be done by hand (Memorandum of Agreement between Advisory Council on Historic Preservation, U.S. Nuclear Regulatory Commission, State Historic Preservation Officer for Georgia and Georgia Power Company).

4.2.3.3 Cultural Properties Along Transmission Line Corridors

Routine maintenance activities in these areas will be in accordance with the Final Cultural Resource Treatment Plans.

4.2.4 Noise Monitoring

Complaints received by Georgia Power Company regarding noise along the high voltage transmission lines (VEGP-Goshen, VEGP-Scherer, VEGP-Thalman, and Georgia side of VEGP-SCEG) and a report of the actions taken in response to any complaints shall be submitted to the NRC staff in the annual report.

5.0 Administrative Procedures

5.1 Review and Audit

The licensee shall provide for review and audit of compliance with the EPP. The audits shall be conducted independently of the individual or groups responsible for performing the specific activity. A description of the organization structure utilized to achieve the independent review and audit function and results of the audit activities shall be maintained and made available for inspection.

5.2 Records Retention

Records and logs relative to the environmental aspects of station operation shall be made and retained in a manner convenient for review and inspection. These records and logs shall be made available to NRC on request.

Records of modifications to station structures, systems and components determined to potentially affect the continued protection of the environment shall be retained for the life of the station. All other records, data and logs relating to this EPP shall be retained for five years or, where applicable, in accordance with the requirements of other agencies.

5.3 Changes in Environmental Protection Plan

Requests for changes in the EPP shall include an assessment of the environmental impact of the proposed change and a supporting justification. Implementation of such changes in the EPP shall not commence prior to NRC approval of the proposed changes in the form of a license amendment incorporating the appropriate revision to the EPP.

5.4 Plant Reporting Requirements

5.4.1 Routine Reports

An Annual Environmental Operating Report describing implementation of this EPP for the previous year shall be submitted to the NRC prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following issuance of the operating license.

The report shall include summaries and analyses of the results of the environmental protection activities required by Subsection 4.2 of this EPP for the report period, including a comparison with related preoperational studies, operational controls (as appropriate), and previous nonradiological environmental monitoring reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of trends toward irreversible damage to the environment are observed, the licensee shall provide a detailed analysis of the data and a proposed course of mitigating action.

The Annual Environmental Operating Report shall also include:

(1) A list of the EPP noncompliances and the corrective actions taken to remedy them.

(2) A list of all changes in station design or operation, tests, and experiments made in accordance with Subsection 3.1 which involved a potentially significant unreviewed environmental question.

(3) A list of nonroutine reports submitted in accordance with Subsection 5.4.2.

In the event that some results are not available by the report due date, the report shall be submitted noting and explaining the missing results. The missing results shall be submitted as soon as possible in a supplementary report.

5.4.2 Nonroutine Reports

A written report shall be submitted to the NRC within 30 days of occurrence of a nonroutine event. The report shall (a) describe, analyze, and evaluate the event, including extent and magnitude of the impact, and plant operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to correct the report event, (d) indicate the corrective

action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (e) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such report at the same time it is submitted to the other agency.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator ~~of an unisolated reactor coolant loop~~ shall be OPERABLE with lift settings as specified in Table ~~3.7-3~~ 3.7-1.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

Replace with insert for 3/4 7-1.

~~a. With (n) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b. With (n-1) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves associated with an operating loop inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~c. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5.

Insert for Page 3/4 7-1

MODE 1:

a. With 4 reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODE 1 may proceed provided that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-2; otherwise reduce thermal power to less than 5% of RATED THERMAL POWER within the next 2 hours.

b. The provisions of Specification 3.0.4 are not applicable.

MODES 2 and 3:

a. With no main steam line Code safety valves in each steam generator OPERABLE, either restore at least one main steam line Code safety valve to OPERABLE status in each steam generator within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

TABLE 3.7-~~3~~ 1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER					LIFT SETTING (± 1%)*	ORIFICE SIZE
	SG-1	SG-2	SG-3	SG-4		
1.	PSV3001	PSV3011	PSV3021	PSV3031	1185 psig	16.0 in.^{2}
2.	PSV3002	PSV3012	PSV3022	PSV3032	1200 psig	16.0 in.^{2}
3.	PSV3003	PSV3013	PSV3023	PSV3033	1210 psig	16.0 in.^{2}
4.	PSV3004	PSV3014	PSV3024	PSV3034	1220 psig	16.0 in.^{2}
	PSV3005	PSV3015	PSV3025	PSV3035	1235 psig	16.0 in.^{2}

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

TABLE 3.7-~~1~~ 2

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING ~~N~~ 4 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	[87]
2	~~[64]~~ 65
3	~~[42]~~ 43

~~TABLE 3.7-2~~

~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING N-1 LOOP OPERATION~~

~~MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR*~~	~~MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)~~
~~1~~	~~[52]~~
~~2~~	~~[38]~~
~~3~~	~~[25]~~

~~*At least two safety valves shall be OPERABLE on the non-operating steam generator~~.

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator auxiliary feedwater pumps and associated flow paths shall be OPERABLE with:

a. Two motor-driven auxiliary feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine-driven auxiliary feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one auxiliary feedwater pump inoperable, restore the required auxiliary feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two auxiliary feedwater pumps inoperable, be in at least HOT STANDBY witin 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1630 psig at a flow of greater than or equal to 175 gpm; (PI-5140, PI-5141, FI-15101, FI-15102, PI-5128, PI-5129)

2) Verifying that the steam turbine-driven pump develops a discharge pressure of greater than or equal to 1655 psig at a flow of greater than or equal to 160 gpm when the secondary steam supply pressure is greater than 978 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position; and

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is ~~placed in automatic control or when above 10% RATED THERMAL POWER~~. in standby for auxiliary feedwater automatic initiation or when above 10 % RATED THERMAL POWER.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal.

~~4.7.1.2.2* An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to each steam generator~~.

~~*This is applicable only for plants that do not use auxiliary feedwater for STARTUP/SHUTDOWN operations~~.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 ~~The~~ Condensate storage tank (CST) V4001 shall be OPERABLE with a contained water volume of at least 340,000 gallons of water. (68.82%)(LI-5101)

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With ~~the~~ CST V4001 inoperable, within 4 hours either:

a. Restore ~~the~~ CST V4001 to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of CST V4002 ~~the [alternate water source]~~ as a backup supply to the auxiliary feedwater pumps and restore the CST V4001 to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 ~~The~~ CST V4001 shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

4.7.1.3.2 CST V4002 ~~The [alternate water source]~~ shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is at least 340,000 gallons (68.82%) (LI-5104), ~~[method dependent upon alternate source]~~ whenever it ~~the [alternate water source]~~ is the supply source for the auxiliary feedwater pumps.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination*	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

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*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than ~~20~~ 14 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

systems (consisting of a main steam isolation valve (MSIV) and its associated bypass valve (MSIBV) per steam line.

3.7.1.5 ~~Each~~ Two main steam line isolation ~~valve (MSIV)~~ systems (consisting of a main steam isolation valve (MSIV) and its associated bypass valve (MSIBV) per steam line. shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

~~MODE 1~~:

~~With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours~~.

~~MODES 2 and 3~~:

~~With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours~~.

Replace with insert to page 3/4 7-9.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV and MSIBV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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* An OPERABLE main steam line isolation system can consist of an OPERABLE MSIV and an inoperable but closed associated MSIBV provided the inoperable MSIBV is maintained closed.

Insert for Page 3/4 7-9

MODE 1:

a. With two main steam line isolation systems in any steam line inoperable; POWER OPERATION may/continue provided each MSIV in the affected steam line is open and at least one main steam line isolation system in the affected steam line is restored to OPERABLE status within 72 hours. Otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

b. With one main steam line isolation system inoperable, power operation may continue provided the MSIV in the affected isolation system is open and the inoperable system is restored to OPERABLE status within 7 days. Otherwise, reduce power to less than or equal to 5% of RATED THERMAL POWER within 2 hours.

MODES 2 and 3:

a. With two main steam line isolation systems in any steam line inoperable, subsequent operation in MODES 2 or 3 may proceed provided at least one main steam line isolation system in the affected steam line is maintained closed. The provisions of Specification 3.0.4 are not applicable. Otherwise, be in HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. With one main steam line isolation system inoperable but closed, subsequent operation in MODES 2 or 3 may proceed provided that the isolation system in the affected steam line is maintained closed. The provisions of Specification 3.0.4 are not applicable.

c. With one main steam line isolation system inoperable but open, either close the OPERABLE isolation system or restore the inoperable system to OPERABLE status within 7 days. Otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the reactor and secondary coolants in the steam generators shall be greater than {70}°F when the pressure of either coolant in the steam generator is greater than {200} psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to {200} psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of the overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its temperatures above 200°F.

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generator shall be determined to be less than {200} psig at least once per hour when the temperature of either the reactor or secondary coolant is less than {70}°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water ~~loops~~ trains shall be OPERABLE ~~.~~ with at least two pumps per train.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one component cooling water ~~loop~~ train OPERABLE, restore at least two ~~loops~~ trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water ~~loops~~ trains shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve ~~(manual, power-operated, or automatic) servicing safety-related equipment~~ that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that ~~:~~

~~1) Each automatic valve servicing safety-related equipment actuates to its correct position on a _______ test signal, and~~

~~2)~~ Each Component Cooling Water System pump starts automatically on a ________ test signal.

Safety Injection

PLANT SYSTEMS

3/4.7.4 ~~SERVICE WATER SYSTEM~~ NUCLEAR SERVICE COOLING WATER (NSCW) SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent ~~service water loops~~ NSCW trains shall be OPERABLE~~.~~ with at least two pumps per train.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one ~~service water loop~~ NSCW train OPERABLE, restore at least two ~~loops~~ trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two ~~service water loops~~ NSCW trains shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. At least once per 18 months during shutdown, by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a Safety Injection test signal, and

2) Each Service Water System pump starts automatically on a Safety Injection test signal.

3/4.7.5 ULTIMATE HEAT SINK ~~[OPTIONAL]~~

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

Replace with insert for 3/4 7-13

~~a. A minimum water level at or above elevation ____ Mean Sea Level, USGS datum, and~~

~~b. An average water temperature of less than or equal to ____°F.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5 The ultimate heat sink shall be determined OPERABLE: ~~at least once per 24 hours by verifying the average water temperature and water level to be within their limits.~~

a. At least once per 24 hours by verifying the average water level and water temperature to be within their limits.

b. At least once per 31 days by verifying that the minimum required number of fans start and operate for at least 15 minutes.

Insert for Page 3/4 7-13

a. Two OPERABLE Nuclear Service Cooling Water (NSCW) tower basins each with:

    1. A minimum water level (LI-1606 and LI-1607) at or above plant elevation of 217' 3" (73% of span).

    2. A maximum water temperature (TE-1642 and TE-1643) of 90°F.

b. Two OPERABLE trains of NSCW tower fans and spray cells. The required number of fans and spray cells per train is a function of ambient wet bulb temperature and shall be in accordance with the following:

Ambient Wet Bulb Temperature	Minimum Required OPERABLE Fans	Minimum Required OPERABLE Fans
≥55°F	4	4
Between 40°F and 60°F	3	3*
≤40°F	2	2*
Between 15°F and 55°F	3	4
≤15°F	2	4

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* Spray required to the cells with OPERABLE fans and spray to other cells isolated.

PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION [OPTIONAL*]

LIMITING CONDITION FOR OPERATION

3.7.6 Flood protection shall be provided for all Safety-Related Systems, components, and structures when the water level of the _____ [usually the ultimate heat sink] exceeds ______ Mean Sea Level, USGS datum, at _________.

APPLICABILITY: At all times.

ACTION:

With the water level at ______ above elevation _____ Mean Sea Level, USGS datum:

a. [Be in at least HOT STANDBY within 6 hours and in at least COLD SHUTDOWN within the following 30 hours], and

b. Initiate and complete within _____ hours, the following flood protection measures:

1. [Plant dependent], and

2. [Plant dependent].

SURVEILLANCE REQUIREMENTS

4.7.6 The water level at ____ shall be determined to be within the limits by:

a. Measurement at least once per 24 hours when the water level is below elevation _____ Mean Sea Level, USGS datum, and

b. Measurement at least once per 2 hours when the water level is equal to or above elevation _____ Mean Sea Level, USGS datum.

* This specification not required if the facility design has adequate passive flood control protection features sufficient to accommodate the Design Basis Flood identified in Regulatory Guide 1.59, August 1973.

initiate and maintain operation of the remaining OPERABLE Control Room Emergency Filtration System in the Emergency mode. Otherwise,

3/4.7.~~7~~ 6 CONTROL ROOM EMERGENCY ~~AIR CLEANUP~~ FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~7~~ 6 Two independent Control Room Emergency ~~Air Cleanup~~ Filtration Systems shall be OPERABLE.

APPLICABILITY: ~~All MODES.~~ MODES 1, 2, 3, and 4. MODES 5 and 6 during movement of irradiated fuel or movement of loads over irradiated fuel.

ACTION:

MODES 1, 2, 3 and 4:

With one Control Room Emergency ~~Air Cleanup~~ Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6: (During movement of irradiated fuel or movement of loads over irradiated fuel.)

a. With one Control Room Emergency ~~Air Cleanup~~ Filtration System inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE Control Room Emergency ~~Air Cleanup~~ Filtration System in the ~~recirculation~~ emergency mode.

b. With both Control Room Emergency ~~Air Cleanup~~ Filtration Systems inoperable, or with the OPERABLE Control Room Emergency ~~Air Cleanup~~ Filtration System, required to be in the ~~recirculation~~ emergency mode by ACTION a., not capable of being powered by an OPERABLE emergency power source, suspend all operations involving ~~CORE ALTERATIONS or positive reactivity changes~~. movement of irradiated fuel or movement of loads over irradiated fuel.

SURVEILLANCE REQUIREMENTS

4.7.~~7~~ 6 Each Control Room Emergency ~~Air Cleanup~~ Filtration System shall be demonstrated OPERABLE:

~~a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to [80]°F;~~

a ~~b.~~ At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters ~~operating;~~ control circuit energized.

SURVEILLANCE REQUIREMENTS (Continued)

b. ~~c.~~ At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

Replace with insert to 3/4 7-15, item 1.

~~1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52~~, ~~Revisions 2, March 1978, and the system flow rate is~~ ________ cfm ± 10%;

Replace with insert to 3/4 7-15, item 2.

~~2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52~~, ~~Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and~~

3) Verifying a system flow rate of 25,000 cfm ± 10% during system operation when tested in accordance with Section 8 of ANSI N510-~~1975~~ 1980.

Replace with insert to 3/4 7-15, item C.

~~d. After every 720 hours of charcoal adsorber operation, by verifying~~, ~~within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978~~, ~~meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%~~;

d. ~~e.~~ At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than ~~[~~6~~]~~ inches Water Gauge while operating the system at a flow rate of 25,000 cfm ± 10%;

2) Verifying that on a Control Room Isolation ~~Containment Phase "A" Isolation and High Smoke Density~~ Test signal, the system automatically switches into a emergency ~~recirculation~~ mode of operation with flow through the HEPA filters and charcoal adsorbers ~~banks;~~

3) Verifying that the system maintains the control room at a positive pressure of greater than or equal to ~~[~~1/8~~]~~ inch Water Gauge ~~at less than or equal to a pressurization flow of ____ cfm relative to adjacent areas during system operation;~~ relative to the outside atmosphere during system operation.

4) Verifying that the heaters dissipate 118 ± 6 kW when tested in accordance with Section 14 of ANSI N510-~~1975~~ 1980; and

5) Verifying that on a Control Room ~~High Chlorine~~/Toxic Gas isolation Test signal, ~~the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks within [15] seconds~~. the control room isolation dampers close within 6 seconds.

Insert to Page 3/4 7-15, item 1

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria of greater than or equal to 99.5% filter efficiency while operating the system at a flow rate of 25,000 cfm $\pm$ 10% and performing the following tests:

   (a) A visual inspection of the control room emergency air cleanup system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

   (b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

   (c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

Insert to 3/4 7-15, item 2.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 80°C and 70% relative humidity.

Insert to 3/4 7-15, item c

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% when tested with methyl iodide at 80°C and 70% relative humidity.

SURVEILLANCE REQUIREMENTS (Continued)

Replace with insert to Page 3/4 7-16

~~f. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of ______ cfm ± 10%; and~~

~~g. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of ______ cfm ± 10%.~~

~~*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)~~

~~**Value applicable will be determined by the following equation:~~

~~$P = \frac{100\% - E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).~~

Insert to Page 3/4 7-16

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.5% of the DOP when they are tested in-place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 25,000 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.5% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 25,000 cfm ±10%.

3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ECCS Pump Room Exhaust Air Cleanup Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one ECCS Pump Room Exhaust Air Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8 Each ECCS Pump Room Exhaust Air Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is _______ cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2 March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of ____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ___ cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal,

3) Verifying that the system maintains the ECCS pump room at a negative pressure of greater than or equal to [1/8] inch Water Gauge relative to the outside atmosphere,

4) Verifying that the filter cooling bypass valves can be manually opened, and

5) Verifying that the heaters dissipate _____ ± ______ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _______ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

3/4.7.7 PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7 Two independent ECCS Piping Penetration Area Filtration and Exhaust Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With one ECCS Piping Penetration Area Filtration and Exhaust System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7 Each ECCS Piping Penetration Area Filtration and Exhaust System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heater control circuit energized.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria of greater than or equal to 99.5% filter efficiency while operating the system at a flow rate of 16,000 cfm $\pm$10% and performing the following tests:

      (a) A visual inspection of the piping penetration area filtration and exhaust system shall be made before each DOP test or activated carbon adsorber section leak test in accordance with Section 5 of ANSI N510-1980.

      (b) An in-place DOP test for the HEPA filters shall be performed in accordance with Section 10 of ANSI N510-1980.

      (c) A charcoal adsorber section leak test with a gaseous halogenated hydrocarbon refrigerant shall be performed in accordance with Section 12 of ANSI N510-1980.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% efficiency when tested with methyl iodide at 80°C and 70% relative humidity.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 16,000 cfm ±10% during system operation when tested in accordance with Section 8 of ANSI N510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Section 13 of ANSI N510-1980 meets the laboratory testing criterion of greater than or equal to 99.8% efficiency when tested with methyl iodide at 80°C and 70% relative humidity.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks of less than 6 inches Water Gauge while operating the system at a flow rate of 16,000 cfm ±10%.

2. Verifying that the system starts on a Containment Ventilation Isolation test signal.

3. Verifying that the heaters dissipate 80±4 kw when tested in accordance with Section 14 of ANSI N510-1980.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.5% of the DOP when they are tested in-place in accordance with Section 10 of ANSI N510-1980 while operating the system at a flow rate of 16,000 cfm ±10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.5% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with Section 12 of ANSI N510-1980 while operating the system at a flow rate of 16,000 cfm ±10%.

3/4.7.~~9~~8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.~~9~~8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.~~9~~8 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type [on any system] are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection [of that system] shall be performed at the first refueling outage. Otherwise, subsequent visual inspections [of a given system] shall be performed in accordance with the following schedule:

No. of Inoperable Snubbers of Each Type [on Any System] per Inspection Period	Subsequent Visual Inspection Period* **
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval for each type of snubber [on a given system] shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found [on that system].

**The provisions of Specification 4.0.2 are not applicable.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type [on that system] that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.~~9~~f. 8 All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers. [For those snubbers common to more than one system, the OPERABILITY of such snubbers shall be considered in assessing the surveillance schedule for each of the related systems.]

d. Transient Event Inspection

An inspection shall be performed of all snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within 6 months following such an event. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using at least one of the following: (1) manually induced snubber movement; or (2) evaluation of in-place snubber piston setting; or (3) stroking the mechanical snubber through its full range of travel.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers of each type shall be tested using one of the following sample plans. The sample plan for each type shall be selected prior to the test period and cannot be changed during the test period. The NRC Regional Administrator shall be notified in writing of the sample plan selected for each snubber type prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.~~9~~f., an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or
8

e. Functional Tests (Continued)

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.~~9~~8f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested; or

3) An initial representative sample of 55 snubbers shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, 1 + C/2, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation N = 55(1 + C/2). Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all the snubbers of that type have been tested.

Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

SURVEILLANCE REQUIREMENTS (Continued)

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) For mechanical snubbers, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.8e. for snubbers not meeting the functional test acceptance criteria.

SURVEILLANCE REQUIREMENTS (Continued)

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test results shall be tested to meet the functional test criteria before installation in the unit. Mechanical snubbers shall have met the acceptance criteria subsequent to their most recent service, and the freedom-of-motion test must have been performed within 12 months before being installed in the unit.

i. Snubber Service Life Program

The service life of hydraulic and mechanical snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.3.

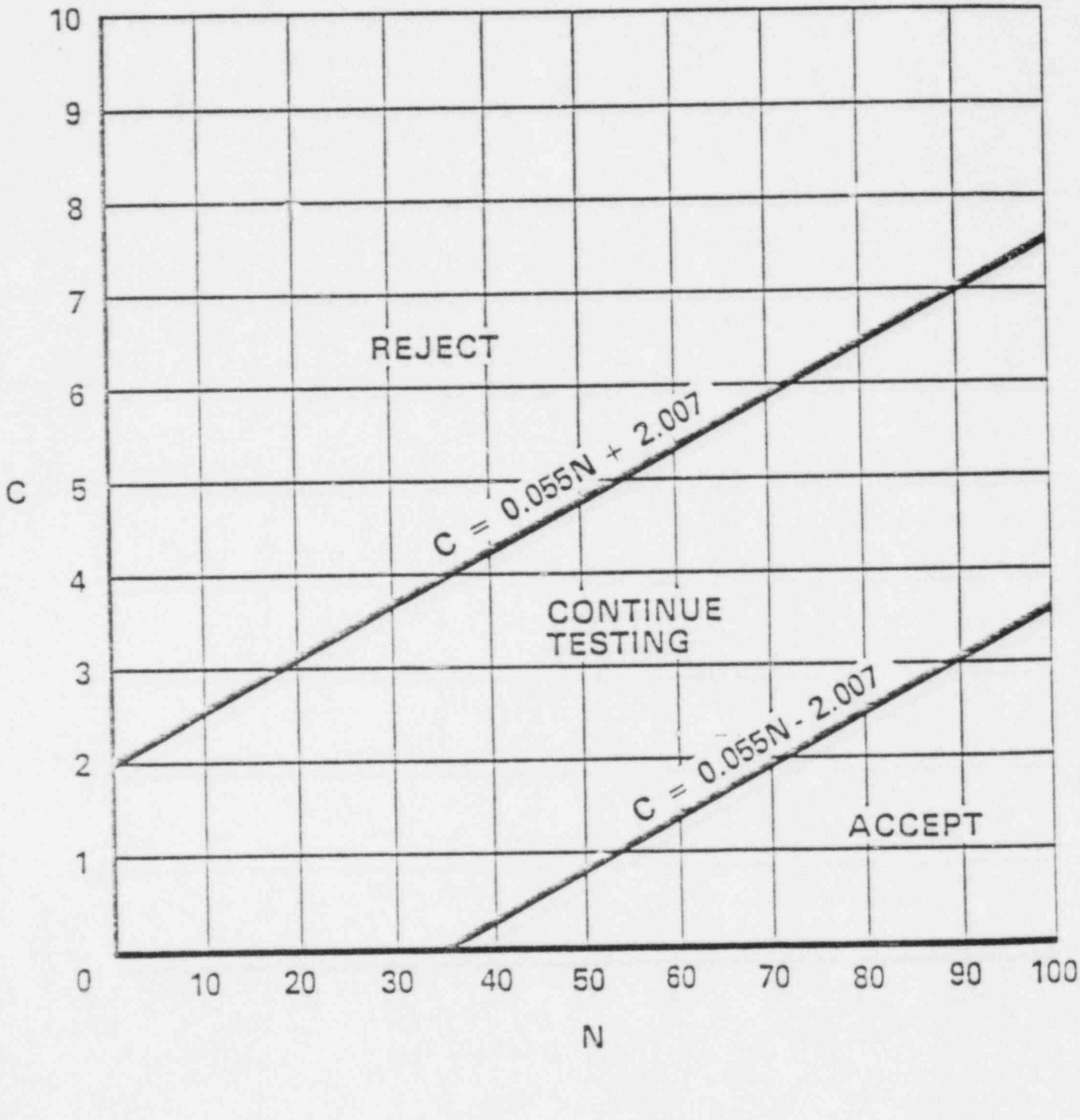

FIGURE 4.7-1

SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST

PLANT SYSTEMS

3/4.7.~~10~~9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.~~10~~9 Each sealed source containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 5 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

   1. Decontaminate and repair the sealed source, or

   2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.~~10~~9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.~~10~~9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

   1) With a half-life greater than 30 days (excluding Hydrogen 3), and

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

~~4.7.10.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination~~.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11.1 The Fire Suppression Water System shall be OPERABLE with:

a. At least [two] fire suppression pumps, each with a capacity of [2500] gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of _____ gallons, and

c. An OPERABLE flow path capable of taking suction from the _____ tank and the ____ tank and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.5, and 3.7.11.6.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the Fire Suppression Water System otherwise inoperable, establish a backup Fire Suppression Water System within 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.11.1.1 The Fire Suppression Water System shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume,

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

c. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

d. [At least once per 6 months by performance of a system flush,]

e. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1) Verifying that each automatic valve in the flow path actuates to its correct position,

   2) Verifying that each pump develops at least [2500] gpm at a system head of [250] feet,

   3) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel, and

   4) Verifying that each fire suppression pump starts [sequentially] to maintain the Fire Suppression Water System pressure greater than or equal to ___ psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least ___ gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975 is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months, during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.11.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The electrolyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

c. At least once per 18 months by verifying that:

1) The batteries, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration, and

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. [Plant dependent - to be listed by name and location.]

b.

c.

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

1) By performing a system functional test which includes simulated automatic actuation of the system, and:

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a __________ test signal, and

b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The following High Pressure and Low Pressure CO_2 Systems shall be OPERABLE:

a. [Plant dependent - to be listed by name and location.]

b.

c.

APPLICABILITY: Whenever equipment protected by the CO_2 Systems is required to be OPERABLE.

ACTION:

a. With one or more of the above required CO_2 Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 Systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position.

4.7.11.3.2 Each of the above required Low Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the CO_2 storage tank level to be greater than ____ and pressure to be greater than ____ psig, and

b. At least once per 18 months by verifying:

1) The system, including valves, associated ventilation system fire dampers, and fire door release mechanisms, actuates manually and automatically upon receipt of a simulated actuation signal, and

2) Flow from each nozzle during a "Puff Test."

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.3.3 Each of the above required High Pressure CO_2 Systems shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying the CO_2 storage tank weight to be at least 90% of full charge weight, and

b. At least once per 18 months by:

1) Verifying the system, including associated ventilation system fire dampers and fire door release mechanisms, actuates manually and automatically upon receipt of a simulated actuation signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The following Halon Systems shall be OPERABLE:

a. [Plant dependent - to be listed by name and location.]

b.

c.

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the above required Halon Systems shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 6 months by verifying Halon storage tank weight to be at least 95% of full charge weight [or level] and pressure to be at least 90% of full charge pressure, and

c. At least once per 18 months by:

1) Verifying the system, including associated Ventilation System fire dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.5 The fire hose stations given in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-4 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.5 Each of the fire hose stations given in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 18 months, by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and re-racking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK NUMBER

*List all fire hose stations required to ensure the OPERABILITY of safety-related equipment.

PLANT SYSTEMS

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.6 The yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11.6 Each of the yard fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

b. At least once per 6 months (once during March, April, or May and once during September, October, or November), by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged, and

c. At least once per 12 months by:

   1) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

   2) Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

   3) Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT NUMBER

*List all Yard Fire Hydrants and Hydrant Hose Houses required to ensure the OPERABILITY of safety-related equipment.

3/4.7.12 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire rated assemblies (walls, floor/ceilings, cable tray enclosures, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by performing a visual inspection of:

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. The OPERABILITY of the fire door supervision system for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days,

b. That each locked closed fire door is closed at least once per 7 days,

c. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours, and a functional test is performed at least once per 18 months, and

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

3/4.7.~~13~~10 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.~~13~~10 The temperature of each area shown in Table 3.7-~~6~~3 shall not be exceeded for more than 8 hours or by more than 30°F.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

a. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-~~6~~3 for more than 8 hours, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate the continued OPERABILITY of the affected equipment. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With one or more areas exceeding the temperature limit(s) shown in Table 3.7-~~6~~3 by more than 30°F, prepare and submit a Special Report as required by ACTION a. above and within 4 hours either restore the area(s) to within the temperature limit(s) or declare the equipment in the affected area(s) inoperable.

SURVEILLANCE REQUIREMENTS

4.7.~~13~~10 The temperature in each of the areas shown in Table 3.7-~~6~~3 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-6

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1.	
2.	
3.	
4.	
5.	

(Later)

3/4 7.11 ENGINEERED SAFETY FEATURES (ESF) ROOM COOLER AND SAFETY-RELATED CHILLER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 Two independent ESF room cooler and safety-related chiller trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one ESF room cooler and safety-related chiller train OPERABLE, restore two trains to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.11 Two ESF room cooler and safety-related chiller trains shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) servicing safety-related equipment that is not locked, sealed or otherwise secured in position, is in its correct position.

b. At least once per 18 months by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a safety injection test signal.

c. At least once per 18 months by verifying that each ESF room cooler fan and each train of safety-related chillers (pump and chiller) start automatically on a safety injection test signal.

3/4.7.12 REACTOR COOLANT PUMP THERMAL BARRIER COOLING WATER ISOLATION

LIMITING CONDITION FOR OPERATION

3.7.12 The reactor coolant pump thermal barrier isolation function shall be OPERABLE.

APPLICABILITY: Modes 1, 2, 3, and 4.

ACTION:

With the reactor coolant pump thermal barrier isolation function inoperable, restore to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.12 The reactor coolant pump thermal barrier isolation function shall be demonstrated OPERABLE by:

a. Verifying that valve HV-2041 automatically closes on thermal barrier outlet high header pressure and high header flow test signals at least once per 36 months on a STAGGERED TEST BASIS.

b. Verifying that valves HV-19051, 19053, 19055, and 19057 automatically close on thermal barrier outlet high flow (FT-19052, 19054, 19056, and 19058) test signals at least once per 18 months.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.7

3/4.7:

Instrument tag numbers were added as appropriate to assist the operator in ensuring compliance with these Technical Specifications.

3.7.1.1:

The deletion of "of an unisolated reactor coolant loop" in the LCO reflects the fact that RCS loop isolation valves are not included in the VEGP design and the deletion of STS Table 3.7-2 is based on the fact that N-1 operation will not be permitted.

Similarly, Action Statement b was deleted to reflect the intent of the plant not to operate in N-1 loop operation.

The revised action statements also allow operation in Modes 2 and 3 provided at least one safety valve is OPERABLE without resetting reactor trip setpoints. Resetting trip setpoints in these modes is not necessary since the low power setpoint will be active. Also, forcing the plant to Cold Shutdown is unnecessary since the specification is applicable in Modes 1, 2, and 3 only. The revised specification correctly replaces Cold Shutdown with Hot Shutdown.

This revision to Specification 3/4.7.1.1 is sponsored to the NRC by the Westinghouse Owners Group in letters OG-152 (June 1985) and OG-161 (October 1985).

4.7.1.2.1.a.4:

This surveillance requirement was revised for clarification. The words "automatic control" imply that the AFWS is controlled automatically as is the main feedwater system. When the AFW pump controls are in the automatic position, the pumps will auto-start on an initiation signal. Once the pumps have started, AFW flow is manually regulated to maintain the desired steam generator water level. See subsection 10.4.9 of the VEGP FSAR.

4.7.1.2.2:

This surveillance requirement was deleted on the basis that the auxiliary feedwater system is used for startup and shutdown operation. See paragraph 10.4.9.2.3 of the VEGP FSAR.

3/4.7.1.3:

See subsection 9.2.6 of the VEGP FSAR for a description of the condensate storage tanks.

4.7.1.4; Table 4.7-1; Footnote *:

See the justification for revisions to Definition 1.10.

3/4.7.1.5:

This specification has been revised to reflect plant design. The VEGP is equipped with redundant main steam isolation valves and associated bypass isolation valves in each steam line. In the case where both MSIVs in a single steam line are inoperable, an outage time of 72 hours (as opposed to 4 hours) was chosen on the strength of an analysis which indicated that the increase in probability of core damage due to a main steam line break was negligible (1.01×10^{-6} for operable MSIVs and 1.02×10^{-6} for two inoperable MSIVs in a single steam line for 72 hours per year). In the case where one MSIV is inoperable for 7 days in a year, the probability of core damage in the event of a main steam line break was found to be 1.02×10^{-6}. For comparison purposes, the probability of core damage in the case of both MSIVs in a single line inoperable for a full year was found to be 2.51×10^{-6}. The action statement for Mode 1 was revised to reduce power to less than or equal to 5-percent of rated thermal power within 2 hours to allow the operator to go to the lesser mode of startup which is addressed in the action associated with Modes 2 or 3. See paragraph 10.3.2.2.4 of the VEGP FSAR for a description of the main steam isolation valves.

3/4.7.3:

The word "loop" was replaced with "train" to be consistent with VEGP specific terminology used in the FSAR. Also, each train includes three 50-percent-capacity pumps with only two of the three pumps required for an operable train.

The surveillance requirements were modified to reflect the fact that the design of the component cooling water system does not include any automatic valves servicing safety-related equipment. See subsection 9.2.2 of the VEGP FSAR.

3/4.7.4:

This specification was revised to refer to "trains" instead of loops to be consistent with terminology used in the FSAR and to reflect that the nuclear service cooling water (NSCW) system includes three 50-percent-capacity pumps per train. Therefore, only two pumps are required for an operable train. See subsection 9.2.1 of the VEGP FSAR.

3/4.7.5:

The terminology was revised to reflect the VEGP design, which consists of two NSCW tower basins, the combined inventory of which comprises the VEGP ultimate heat sink. Also, the water level is specified in plant elevation which does not necessarily correspond to feet above mean sea level. See subsection 9.2.5 of the VEGP FSAR.

The specified maximum water temperature is not an average temperature but rather that which is read in the common pump discharge header, and thus is a true measure of the temperature of the water being supplied to the cooled components.

Operability as a function of available NSCW tower fans and ambient wet bulb temperature is based on calculations which demonstrate that, given the specified ambient wet bulb temperature limits, the specified number of fan and spray cells are the minimum required to ensure sufficient cooling capacity for safe shutdown. This allows operational flexibility such that, should tower maintenance be required (fill replacement, fan maintenance, etc.) operation could continue.

3/4.7.6 (STS):

This specification was deleted in its entirety since the VEGP design includes adequate passive flood control protection in conformance with Regulatory Guide 1.59. See paragraph 1.9.59.2 of the VEGP FSAR.

3.7.6:

Applicability and Actions:

Our draft involves two deviations from the STS: (1) The applicability for Modes 5 and 6 was revised such that the control room emergency filtration system is required to be OPERABLE only during movement of irradiated fuel or movement of loads over irradiated fuel; (2) The action for Modes 1, 2, 3, and 4 was revised to allow the option of initiating and maintaining the other filtration train in the emergency mode. Action b under Modes 5 and 6 was also revised to require suspension of operations involving movement of irradiated fuel and loads over irradiated fuel, to be consistent with the applicability as revised.

The justification for the revision to the action for Modes 1, 2, 3 and 4 has its basis in the fact that if the OPERABLE filtration train is placed in the emergency mode and maintained in that mode, the operators will be protected in the event of an accident or until the inoperable train can be restored to operable status.

The justification for the revision to the applicability during Modes 5 and 6 has its basis in the fact that in these modes the filtration system is only necessary to protect the operators against a release resulting from a fuel

handling accident. Therefore, as long as there is no movement of irradiated fuel or loads over irradiated fuel, the system should not be required to be operable. Similarly, Action b was revised to be consistent with the revised applicability.

The modifications to the applicability statement have been approved and implemented in the Farley Technical Specifications.

4.7.6:

These surveillance requirements have been revised to incorporate plant-specific features and improvements which have already been approved by the NRC and implemented in the Farley Nuclear Plant Technical Specifications. See subsection 6.4.2 of the VEGP FSAR for a description of the control room emergency filtration system.

4.7.6.a (STS):

Surveillance Requirement 4.7.6.a was deleted from this specification and moved to 4.7.12. As discussed in the bases, the operability requirement ensures that the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system. Because the control room emergency filtration system is normally not in operation it does not ensure that the ambient air temperature does not exceed the allowable. In addition, as written, if the control room temperature exceeded the allowable, the emergency system must be declared inoperable, while it may be the means to reinstate proper temperature.

4.7.6.a:

Filtration unit heaters are designed to maintain the moisture content of the air entering the charcoal adsorption bed below 70 percent. The heaters are enabled by a contact in the fan starting circuit and heater control is governed by a moisture sensor in the heater outlet duct. There are no indicating lights associated with the heater.

The surveillance requirement states that the heaters must remain energized during the entire surveillance test period. With no indication of heater status this has necessitated the addition of steps in the operations surveillance procedures which require the measurement of voltage to the heaters. This is impractical from a safety standpoint and also from the standpoint that the heater is actually under control of the moisture sensor and will cycle on and off during the test. If the heater is operating correctly, it should not remain energized for the 10 hours required in the Technical Specifications. Additionally, during periods of low relative humidity the heaters may not energize at all. Therefore, the surveillance requirement has been revised to require that the heater control circuit be energized during the 10-hour period. This meets the intent of the specification and is consistent with plant design.

4.7.6.b:

The revisions to this specification are as follows:

1. The surveillance requirement to verify a maximum total bypass flow of 1 percent was deleted on the basis that the VEGP design does not include divert valves or the capability for bypass.

2. References to positions c.5.a, c.5.c, c.5.d, and c.6.b of Regulatory Guide 1.52 were replaced with references to Sections 5, 10, 12, and 13 of ANSI N510-1980. This was done for the purpose of clarification in an effort to reduce the potential for a misunderstanding of the requirements. The 1980 version of ANSI N510 was used as stated in subsection 1.9.52 of the VEGP FSAR.

3 The laboratory testing criterion of 99.8 percent is based on the following equation for methyl iodide penetration which is found in draft Rev. 5 to NUREG-0452:

$$P = \frac{100\% - E}{SF}$$

where P = maximum permissible methyl iodide penetration, percent
E = efficiency assumed in the SER for methyl iodide removal, percent
SF = safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

If E = 99 percent (Table 15.5 of the VEGP SER) and SF = 5 since VEGP is equipped with heaters, then:

$$P = 0.2\%$$

Therefore, the laboratory testing criterion for the efficiency for methyl iodide removal should be 99.8 percent.

The filter efficiency of 99.95 percent is based on the assumption of bypass leakage. The VEGP control room emergency filtration system does not include the capability for bypass and, therefore, a filter efficiency of 99.95 percent is overly restrictive. A less restrictive value of 99.5 percent, when multiplied by the laboratory criterion of 99.8 percent, yields an overall iodine removal efficiency of 99.3 percent which is still greater than the efficiency assumed in the SER. The limiting criterion for the appropriate filter testing efficiency should be to ensure that at the completion of a surveillance interval, the filters will remove at least as much iodine as was assumed in the accident analysis. Therefore, the value of 99.5 percent should be sufficient to ensure that this criterion is met.

Reference to Section 8 of ANSI N510-1980 was added to the surveillance on system flowrate for the purpose of clarification.

4.7.6.c:

Reference to position c.6.b of Regulatory Guide 1.52 was replaced with reference to Section 13 of ANSI N510-1980 for the purpose of clarification. The laboratory testing criteria are listed in the surveillance requirement and the efficiency for methyl iodide removal of 99.8 percent is discussed under 4.7.6.b above.

4.7.6.d:

Item 2 was revised to be plant specific. The signals which result in control room isolation are shown on figure 7.2.1-1 (sheet 8) of the VEGP FSAR.

Item 3 was revised to reflect that the positive pressure for VEGP is measured relative to atmosphere rather than the adjacent areas. The VEGP system provides the required pressurization flow to maintain 1/8 in. Water Guage. This flow may vary depending on such conditions as ambient wind velocities, direction, and temperature. This is consistent with the toxic gas analysis for VEGP.

Item 4 was revised to reference Section 14 of ANSI N510-1980 for clarification.

Item 5 was revised to reflect plant design with regard to toxic gas isolation.

4.7.6.e:

A filter efficiency of 99.5 percent was specified based on the argument presented above in 4.7.6.b and reference to Section 10 of ANSI N510-1980 was inserted for clarification.

4.7.6.f:

A filter efficiency of 99.5 percent was specified based on the argument presented above in 4.7.6.c and reference to Section 12 of ANSI N510-1980 was inserted for clarification.

3/4.7.8, ECCS Pump Room Exhaust Air Cleanup System (STS):

This specification was deleted since the VEGP design does not include such a system. See FSAR paragraph 6.5.1.1 for those systems which are taken credit for. The ESF room coolers are covered by Specification 3/4.7.13.

3/4.7.7. Piping Penetration Area Filtration and Exhaust System:

This specification was added to reflect that credit is taken for the operation of this system to maintain offsite radiation exposures associated with post-accident recirculation outside containment within the guideline valves of 10 CFR 100. See FSAR paragraphs 6.5.1.1 and 9.4.3.2.2.2.

4.7.7.a:

The justification for the revision to this specification is the same as previously presented for 4.7.6.a.

4.7.7.b:

The justification for the revisions to this specification is the same as previously presented for 4.7.6.b.

4.7.7.c:

The justification for the revision to this specification is the same as previously presented for 4.7.6.c.

4.7.7.d:

The requirement that filter cooling bypass valves be verified capable of being manually opened was deleted because filter cooling is not part of the VEGP design.

Reference to Section 15 of ANSI N510-1980 was inserted for clarification.

4.7.7.e:

See the justification provided for 4.7.6.e.

4.7.7.f:

See justification provided for 4.7.6.f.

4.7.10.3 (STS):

This reporting requirement was moved to Section 6.9 in an effort to get the annual reporting requirement in one location within the Technical Specifications. This revision has already been approved by the NRC and implemented in the Farley Nuclear Plant Technical Specifications.

3/4.7.11, Fire Suppression Systems:

3/4.7.12, Fire Rated Assemblies:

See the justification provided for the deletion of Specification 3/4.3.3.8.

3/4.7.11:

This specification was added on the basis that the operation of the ESF room cooler and safety-related chiller system ensures that the ambient air temperature does not not exceed the allowable temperature for continuous duty rating for the equipment cooled by the system. See paragraph 9.2.9.1 of the VEGP FSAR.

3/4.7.12:

This specification was added to ensure the operability of the isolation provisions that prevent a loss of reactor coolant through the nonsafety portions of the ACCW system. See paragraph 7.6.6.4 of the VEGP FSAR.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E Distribution System, and

b. Two separate and independent diesel generators, each with:

1) ~~Separate~~ day ~~and engine-mounted fuel~~ tank~~s~~ containing a minimum volume of 750 gallons of fuel,

2) A separate Fuel Storage System containing a minimum volume of 64,000 gallons of fuel,

3) A separate fuel transfer pump,

~~4) Lubricating oil storage containing a minimum total volume of ___ gallons of lubricating oil, and~~

~~5) Capability to transfer lubricating oil from storage to the diesel generator unit~~.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one offsite circuit of the above-required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If either diesel generator has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirements 4.8.1.1.2.a.~~5~~ 4 and 4.8.1.1.2.a.~~6~~ 5 for each such diesel generator, separately, within 24 hours unless the diesel generator is already operating. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT ~~SHUTDOWN~~ STANDBY within the next ~~12~~ 6 hours and in COLD SHUTDOWN within the following ~~24~~ 30 hours.

b. With either diesel generator inoperable*, demonstrate the OPERABILITY of the above required A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.~~5~~ 4 and

*A diesel generator shall be considered to be inoperable from the time of failure until it satisfies the requirements of Surveillance Requirement 4.8.1.1.2.a.5.

ACTION (Continued)

4.8.1.1.2.a.~~6~~ 5 within 24 hours*.* Restore the inoperable diesel generator to OPERABLE status within 72 hours or be in at least HOT STANDBY ~~SHUTDOWN~~ within the next ~~12~~ 6 hours and in COLD SHUTDOWN within the following ~~24~~ 30 hours.

c. With one diesel generator inoperable in addition to ACTION ~~a.~~ b or ~~b.~~ c. above, verify that:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

2. When in MODE 1, 2, or 3, the steam-driven auxiliary feedwater pump is OPERABLE.

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators separately by performing the requirements of Specification 4.8.1.1.2.a.~~5)~~ 4 and 4.8.1.1.2.a.~~6)~~ 5 within ~~1 hour and at least once per~~ 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Insert 2 ~~With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~.

~~e.~~ f. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing the requirements of Specification 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Insert 3 ~~Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the Onsite Class 1E Distribution System shall be:

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY.

Insert 1

c. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. offsite source by performing Surveillance Requirement 4.8.1.1.2.a.4 within 1 hour and at least once per 8 hours thereafter, and, if the diesel generator became inoperable due to any cause other than preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.8.1.1.2.a.5 within 8 hours, unless the OPERABLE diesel generator is already operating. Restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of 3.8.1.1, Action Statement a or b, as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source. A successful test of diesel generator OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.5 performed under the Action Statement for an OPERABLE diesel generator or a restored to OPERABLE diesel generator satisfies the diesel generator test requirement of Action Statement a or b.

Insert 2

Following restoration of one offsite source, follow Action Statement a with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite a.c. circuit. A successful test(s) of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.5 performed under this Action Statement for the OPERABLE diesels satisfies the diesel generator test requirement for Action Statement a.

Insert 3

Following restoration of one diesel generator unit, follow Action Statement b with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator. A successful test of diesel OPERABILITY per Surveillance Requirement 4.8.1.1.2.a.5 performed under this Action Statement for a restored to OPERABLE diesel satisfies the diesel generator test requirements of Action Statement b.

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, and indicated power availability. ~~and~~

~~b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring (manually and automatically) unit power supply from the normal circuit to the alternate circuit.~~

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1) Verifying the fuel level in the day ~~and engine-mounted fuel~~ tank,

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day ~~and engine-mounted~~ tank,

~~4) Verifying the lubricating oil inventory in storage;~~

4 ~~5~~) Verifying the diesel starts* ~~from ambient condition~~ and ~~accelerates to at least [900] rpm in less than or equal to [10] seconds.*~~ and The generator voltage and frequency ~~shall be~~ are ~~[4160] + [420]~~ 4160 +240, -410 volts and ~~[60]~~ 60 ± ~~[1.2]~~ 1.2 Hz within ~~[10]~~ 11.4 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5 ~~6~~) Verifying the generator is synchronized, loaded to ~~greater than or equal to [continuous rating]~~ an indicated 6100-7000 kW** ~~kW in less than or equal to [60] seconds*~~, and operates ~~with a load greater than or equal to [continuous rating]~~ for at least 60 minutes, and

6 ~~7~~) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

Insert 3A 7) Verifying the pressure in at least one diesel generator air start receiver to be greater than or equal to 210 psig.

---

*These diesel generator starts ~~from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures such as gradual loading (>150 sec) as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.~~

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate the test.

Insert 3A

*All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day ~~and engine-mounted fuel~~ tanks;

space

c. At least once per 92 days ~~[or once per 31 days (if the groundwater table is equal to or higher than the bottom of the fuel oil storage tanks)]~~ by checking for and removing accumulated water from the fuel oil storage tanks;

Insert 4 to 3/4 3-4

~~d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:~~

~~1) A water and sediment content of less than or equal to 0.05 volume percent;~~

~~2) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centostokes;~~

~~3) A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to ( ) but less than or equal to ( ) or an API gravity at 60°F of greater than or equal to ( ) degrees but less than or equal to ( ) degrees;~~

~~4) An impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70; analysis shall be completed within 7 days after obtaining the sample but may be performed after the addition of new fuel oil; and~~

~~5) The other properties specified in Table 1 of ASTM-D975-1977 and Regulatory Guide 1.137, Revision 1, October 1979, Position 2.a., when tested in accordance with ASTM-D975-1977; analysis shall be completed within 14 days after obtaining the sample but may be performed after the addition of new fuel oil.~~

g. ~~e.~~ At least once per 18 months,# during shutdown, by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with ~~its manufacturer's~~ recommendations for this class of standby service;

2) Verifying the diesel generator capability to reject a load of greater than or equal to ~~[largest single emergency load]~~ ≥ 671 kW (motor-driven auxiliary feedwater pump) kW while maintaining voltage at ~~[4160] ± [420]~~ 4160 + 240, -410 volts and ~~frequency at~~ speed of ~~[60] ± [1.2] Hz~~ 484 RPM (less than or equal to 75% of the difference between nominal speed and the Overspeed Trip Setpoint, or nominal speed plus ~~or 15% above nominal whichever is less~~);

#For any start of a diesel pursuant to these surveillance requirements, the diesel must be operated with a load in accordance with the manufacturer's recommendations.

Insert to Page 3/4 8-4

d. By sampling new fuel oil in accordance with ASTM-D4057 prior to addition to storage tanks and:

   1) By verifying in accordance with the tests specified in ASTM-D975-81 prior to addition to the storage tanks that the sample has:

      a) An API Gravity of within 0.3 degrees at 60°F, or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees:

      b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification;

      c) A flash point equal to or greater than 125°F; and

      d) A clear and bright appearance with proper color when tested in accordance with ASTM-D4176-82.

   2) By verifying within 30 days of obtaining the sample that the other properties specified in Table 1 of ASTM-D975-81 are met when tested in accordance with ASTM-D975-81 except that the analysis for sulfur may be performed in accordance with ASTM-D1552-79 or ASTM-D2622-82.

e. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

f. At least once per 184 days by:

   1) Verifying the diesel starts* and the generator voltage and frequency are 4160 +240, -410 volts and 60 ±1.2 Hz within 11.4 seconds after the start signal. The diesel generator shall be started for this test by using one of the signals listed in Surveillance Requirement 4.8.1.1.2.a.4. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.4, may also serve to concurrently meet those requirements as well.

   2) Verifying the generator is synchronized, loaded to an indicated value of 6100 to 7000 kW** in less than or equal to 60 seconds, and operates for at least 60 minutes. This test, if it is performed so it coincides with the testing required by Surveillance Requirement 4.8.1.1.2.a.5, may also serve to concurrently meet those requirements as well.

---

* All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress on the diesel engine.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate the test.

3) Verifying the diesel generator capability to reject a load of ~~[continuous rating]~~ 7000 kW without tripping. The generator voltage shall not exceed ~~[4784]~~ 4760 volts during and following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

   b) Verifying the diesel starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within ~~[10]~~ 11.5 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~[4160] ± [420]~~ 4160 +240, -410 volts and [60] ± [1.2] Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts* on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be ~~[4160] ± [420]~~ 4160+240,-410 volts and [60] ± [1.2] Hz within ~~[10]~~ 11.4 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

   b) Verifying the diesel starts* on the auto-start signal, energizes the emergency busses with permanently connected loads within ~~[10]~~ 11.5 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at [4160] +240, -410 ~~± [420]~~ volts and [60] ± [1.2] Hz during this test; and

   c) Verifying that all automatic diesel generator trips, except engine overspeed, low lube oil pressure, high jacket water temperature, and generator differential, are automatically bypassed. ~~upon loss of voltage on the emergency bus concurrent with a Safety Injection Actuation signal.~~

---

* All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

7) Verifying the diesel generator operates* for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to ~~greater than or equal to [2-hour rating] kW~~ and during the remaining 22 hours of this test, the diesel generator shall be loaded to ~~greater than or equal to [continuous rating] kW~~. The generator voltage and frequency shall be ~~[4160] ± [420]~~ volts and [60] ± [1.2] Hz ~~within [10] seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits~~ during this test.** Within 5 minutes after completing this 24-hour test, perform Specification 4.8.1.1.2e.6)b);***

Insert 5

an indicated 6100-7000 kW **

4160 +240, -410

8) Verifying that the auto-connected loads to each diesel generator do not exceed ~~the 2000-hour rating of~~ 6100 kW;

9) Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10) Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day ~~and engine-mounted~~ tank of each diesel via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

~~13) Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:~~

~~a) [Turning gear engaged], or~~

~~b) [Emergency stop].~~

See insert 6 for footnotes *, **, and ***.

*

\# ~~*~~If Specification 4.8.1.1.~~2e~~ 2g.6)b) is not satisfactorily completed, it is not necessary to repeat the preceding 24-hour test. Instead, the diesel generator may be operated at ~~[continuous rating] kW~~ the load specified by Surveillance Requirement 4.8.1.1.2.a.5 for 1 hour or until operating temperature has stabilized.

**

***

INSERT 5

an indicated target value of 7650 kW (between 7600 and 7700 kW)**

INSERT 6

* All engine starts for the purpose of surveillance testing as required by 4.8.1.1.2 may be preceded by an engine prelube period as recommended by the manufacturer to minimize mechanical stress and wear on the diesel engine.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band or momentary variations due to changing bus loads shall not invalidate the test.

*** Failure to maintain voltage and frequency requirements due to grid disturbances does not render a 24-hour test as a failure.

~~14) Verifying that with all diesel generator air start receivers pressurized to less than or equal to ___ psig and the compressors secured, the diesel generator starts at least [5] times from ambient conditions and accelerates to at least [900] rpm in less than or equal to [10] seconds~~.

~~f. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, during shutdown, and verifying that both diesel generators accelerate to at least [900] rpm in less than or equal to [10] seconds; and~~

h. ~~g.~~ At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution, and

2) Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110% of the system design pressure.

~~4.8.1.1.3 Reports - All diesel generator failures, valid or nonvalid, shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests on a per nuclear unit basis is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977~~.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures in Last 20 Valid Tests*	~~Number of Failures in Last 100 Valid Tests*~~	Test Frequency
$\leq$ 1	~~$\leq$ 4~~	Once per 31 days
$\geq$ 2**	~~$\geq$ 5~~	Once per 7 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, but determined on a per diesel generator basis.

For the purposes of determining the required test frequency, the previous test failure count may be reduced to zero if a complete diesel overhaul to like-new condition is completed, provided that the overhaul, including appropriate post-maintenance ~~operating~~ and testing, is specifically approved by the manufacturer and if acceptable reliability has been demonstrated. The reliability criterion shall be the successful completion of 14 consecutive tests in a single series. Ten of these tests shall be in accordance with the routine Surveillance Requirements 4.8.1.1.2.a.5 and 4.8.1.1.2.a.6 and four tests in accordance with the 184-day testing requirement of Surveillance Requirements 4.8.1.1.2.f ~~4.8.1.1.2.a.5 and 4.8.1.1.2.a.6~~. If this criterion is not satisfied during the first series of tests, any alternate criterion to be used to transvalue the failure count to zero requires NRC approval.

**The associated test frequency shall be maintained until seven consecutive failure free demands have been performed and the number of failures in the last 20 valid demands has been reduced to one.

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the Onsite Class 1E Distribution System, and

b. One diesel generator with:

   1) A fuel day ~~Day and engine-mounted fuel~~ tanks containing a minimum volume of 750 gallons of fuel,

   2) A fuel storage system containing a minimum volume of 64,000 gallons of fuel,

   3) A fuel transfer pump,

   ~~4) Lubricating oil storage containing a minimum total volume of ___ gallons of lubricating oil, and~~

   ~~5) Capability to transfer lubricating oil from storage to the diesel generator unit.~~

APPLICABILITY: MODES 5 and 6.

in accordance with Specification 3.4.9.3.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, ~~depressurize and vent~~ the Reactor Coolant System ~~through a greater than or equal to ___ square inch vent.~~ In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

provide relief capability for

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the requirements of Specifications 4.8.1.1.1, 4.8.1.1.2 (except for Specification 4.8.1.1.2a.~~6~~5)), ~~and 4.8.1.1.3~~.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical sources shall be OPERABLE:

~~a. [250/125]-volt Battery Bank No. 1, and its associated full capacity charger, and~~

~~b. [250/125]-volt Battery Bank No. 2, and its associated full capacity charger~~.

See insert to page 3/4 8-10.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

at least one

both

With one of the required battery banks and/or ~~full-capacity~~ chargers inoperable, restore the inoperable battery bank and/or ~~full-capacity~~ charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~This specification is intended for use on plants with two divisions of D.C. power only. Modifications may be necessary, on a plant-unique basis, to accommodate different designs~~.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each [~~250/~~125]-volt battery bank and one associated charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1) The parameters in Table 4.8-2 meet the Category A limits, and

2) The total battery terminal voltage is greater than or equal to ~~[250/129]~~ 126 volts on float charge.

Insert to Page 3/4 8-10

a. 125 V-dc Battery bank 1AD1B, and one of its associated full-capacity chargers.

b. 125 V-dc Battery bank 1BD1B, and one of its associated full-capacity chargers.

c. 125 V-dc Battery bank 1CD1B, and one of its associated full-capacity chargers.

d. 125 V-dc Battery bank 1DD1B, and one of its associated full-capacity chargers.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below ~~[220/110]~~ 106.2 volts, or battery overcharge with battery terminal voltage above ~~[360/150]~~ 140 volts, by verifying that:

1) The parameters in Table 4.8-2 meet the Category B limits,

2) There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than ~~[150 x 10-6] ohm, and~~ 20 % over the average as measured during the plant preoperational tests, and

3) The average electrolyte temperature of ~~[a representative number] of~~ twelve connected cells is above ~~[60]~~ 55° F.

c. At least once per 18 months by verifying that:

1) The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2) The cell-to-cell and terminal connections are clean, tight, and coated with anticorrosion material,

3) The resistance of each cell-to-cell and terminal connection is less than or equal to ~~[150 x 10-6] ohm, and~~ 20 % over the averages measured during the plant preoperational tests and

4) The battery charger will supply at least ~~[400]~~ amperes for system A and B, 300 amperes for system C, and 200 amperes for system D at nominally ~~[125/250]~~ volts for at least ~~[8]~~ 12 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test;

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60-month interval this performance discharge test may be performed in lieu of the battery service test required by Specification 4.8.2.1d.; and

f. At least once per 18 months, during shutdown, by giving performance discharge tests of battery capacity to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
PARAMETER	LIMITS FOR EACH DESIGNATED PILOT CELL	LIMITS FOR EACH CONNECTED CELL	ALLOWABLE(3) VALUE FOR EACH CONNECTED CELL
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(6)	> ~~2.07~~ 2.10 volts
Specific Gravity(4)	≥ ~~1.200~~(5) 1.195	≥ ~~1.195~~ 1.190	Not more than 0.020 below the average of all connected cells
		Average of all connected cells > ~~1.205~~ 1.200	Average of all connected cells ≥ ~~1.195~~(5) 1.190

TABLE NOTATIONS

(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.

(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.

(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

(4) Corrected for electrolyte temperature and level.

(5) Or battery charging current is less than (2) amps when on float charge.

(6) Corrected for average electrolyte temperature.

D.C. SOURCES

SHUTDOWN

(one train related pair of 125-V battery banks (either 125-V battery banks 1AD1B and 1CD1B or 125-V battery banks 1BD1B and 1DD1B) and one associated full-capacity charger per battery bank.

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, ~~one [250/125]-volt battery bank and its associated full-capacity charger~~ shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

at least one

both

With the required battery bank and/or ~~full-capacity~~ chargers inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel; initiate corrective action to restore the required battery bank and ~~full-capacity~~ charger to OPERABLE status as soon as possible, and within 8 hours, ~~depressurize and vent~~ provide relief capability for the Reactor Coolant System ~~through a ___ square inch vent.~~ in accordance with Specification 3.4.9.3

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required ~~[250/~~125]-volt battery bank and full-capacity charger shall be demonstrated OPERABLE in accordance with Specification 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open [both] between redundant busses within the unit [and between units at the same station]:

a. Division #1 A.C. Emergency Busses consisting of:

1) [4160]-Volt Emergency Bus # ______, and
2) [480]-Volt Emergency Bus # ______.

b. Division #2 A.C. Emergency Busses consisting of:

1) [4160]-Volt Emergency Bus # _____, and
2) [480]-Volt Emergency Bus # _____.

c. [120]-Volt A.C. Vital Bus # _____ energized from its associated inverter connected to D.C. Bus # _____*,

d. [120]-Volt A.C. Vital Bus # _____ energized from its associated inverter connected to D.C. Bus # _____*,

e. [120]-Volt A.C. Vital Bus # _____ energized from its associated inverter connected to D.C. Bus # _____*,

f. [120]-Volt A.C. Vital Bus # _____ energized from its associated inverter connected to D.C. Bus # _____*,

g. [250/125]-Volt D.C. Bus #1 energized from Battery Bank #1, and

h. [250/125]-Volt D.C. Bus #2 energized from Battery Bank #2.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. emergency busses not fully energized, reenergize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. vital bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. bus: (1) reenergize the A.C. vital bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) reenergize the A.C. vital bus from its associated inverter connected to its associated D.C. bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Two inverters may be disconnected from their D.C. bus for up to 24 hours as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided: (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. bus.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With one D.C. bus not energized from its associated battery bank, reenergize the D.C. bus from its associated battery bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical Busses shall be energized in the specified manner.

a. A.C. Emergency Busses consisting of:
    1. Train A
        a) 4160 volt switchgear 1AA02
        b) 480 volt switchgear 1AB04
            1) MCC 1ABE
        c) 480 volt switchgear 1AB05
            1) MCC 1ABA
            2) MCC 1ABC
            3) MCC 1ABF
        d) 480 volt switchgear 1AB15
            1) MCC 1ABB
            2) MCC 1ABD
    2. Train B
        a) 4160 volt switchgear 1BA03
        b) 480 volt switchgear 1BB06
            1) MCC 1BBE
        c) 480 volt switchgear 1BB07
            1) MCC 1BBA
            2) MCC 1BBC
            3) MCC 1BBF
        d) 480 volt switchgear 1BB16
            1) MCC 1BBB
            2) MCC 1BBD
b. 120 volt A.C. vital Busses
    1. Associated with Train A
        a) Channel I
            1) Panel 1AY1A energized from inverter 1AD1I1 connected to switchgear 1AD1*
            2) Panel 1AY2A energized from inverter 1AD1I11 connected to switchgear 1AD1*
        b) Channel III
            1) Panel 1CY1A energized from inverter 1CD1I3 connected to switchgear 1CD1*
    2. Associated with Train B
        a) Channel II
            1) Panel 1BY1B energized from inverter 1BD1I2 connected to switchgear 1BD1*
            2) Panel 1BY2B energized from inverter 1BD1I12 connected to switchgear 1BD1*
        b) Channel IV
            1) Panel 1DY1B energized from inverter 1DD1I4 connected to switchgear 1DD1*
c. 125 volt D.C. Busses consisting of:
    1. Associated with Train A
        a) System A
            1) Switchgear 1AD1 energized from battery 1AD1B
            2) MCC 1AD1M energized from switchgear 1AD1

3) Distribution panel 1AD11 energized from switchgear 1AD1
4) Distribution panel 1AD12 energized from switchgear 1AD1

b) System C
1) Switchgear 1CD1 energized from battery 1CD1B
2) MCC 1CD1M energized from switchgear 1CD1
3) Distribution panel 1CD11 energized from switchgear 1CD1

2. Associated with Train B

a) System B
1) Switchgear 1B11 energized from battery 1BD1B
2) MCC 1BD1M energized from switchgear 1BD1
3) Distribution panel 1BD11 energized from switchgear 1BD1
4) Distribution panel 1BD12 energized from switchgear 1BD1

b) System D
1) Switchgear 1DD1 energized from battery 1DD1B
2) Distribution panel 1DD11 energized from switchgear 1DD1

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

a. With one of the required Trains of A.C. Emergency Busses not fully energized, re-energize the Train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 120 volt A.C. Vital Bus either not energized from its associated inverter or with the inverter not connected to its associated Switchgear: (1) Re-energize the Panel within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) Re-energize the Panel from its associated inverter within 24 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one 125 volt D.C. Bus not energized in the specified manner, re-energize the Switchgear or Distribution Panel in the specified manner within 2 hours or be in at least HOT STANDBY in 6 hours and in COLD SHUTDOWN in the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the specified manner at least once per 7 days by verifying correct breaker alignment and indicated voltage.

* Up to two inverters in a single train may be disconnected from their associated Switchgear for up to 24 hours as necessary, for the purpose of Battery Bank or inverter maintenance provided: (1) Their associated Panels are energized from their regulated transformers and (2) The panels associated with the other Battery Bank powered from that AC Train are energized in the specified manner.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses(*) shall be energized in the specified manner:

a. One ~~division~~ train of A.C. emergency busses consisting of one ~~[~~4160~~]~~-volt switchgear, three ~~and one~~ ~~[~~480~~]~~-volt A.C. ~~emergency bus~~ switchgear, and six 480 volt AC Motor Control Centers.

b. ~~Two~~ One train of ~~[~~120~~]~~-volt A.C. vital busses energized from their associated inverters connected to their respective D.C. busses, and

c. ~~One [250/~~ One train of 125~~]~~-volt D.C. ~~bus~~ switchgear and associated distribution equipment energized from its associated battery bank.

APPLICABILITY MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours, ~~depressurize and vent~~ provide relief capability for the RCS ~~through at least a ____ square inch vents~~ in accordance with Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

* All electrical busses shall be from the same train.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

A.C. CIRCUITS INSIDE PRIMARY CONTAINMENT

LIMITING CONDITION FOR OPERATION

3.8.4.1 At least the following A.C. circuits inside primary containment shall be deenergized:

a. Circuit numbers [___, ___, ___ and ___] in panel [ ].

b. Circuit numbers [___, ___, ___ and ___] in panel [ ].

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above required circuits energized, trip the associated circuit breaker(s) in the specified panel(s) within 1 hour.

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the above required A.C. circuits shall be determined to be deenergized at least once per 24 hours* by verifying that the associated circuit breakers are in the tripped condition.

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*Except at least once per 31 days if locked, sealed, or otherwise secured in the tripped condition.

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES AND FEEDER BREAKERS TO ISOLATION TRANSFORMERS BETWEEN 480 V CLASS 1E BUSSES AND NON-CLASS 1E EQUIPMENT

LIMITING CONDITION FOR OPERATION

3.8.4.~~2~~ 1 All containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V Class 1E busses and non-class 1E equipment ~~given in Table 3.8-1~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective device(s) ~~given in Table 3.8-1~~ and feeder breakers to isolation transformers between 480 V Class 1E busses and non-Class 1E equipment inoperable:

See insert for 3/4 8-17

a. ~~Restore the protective device(s) to OPERABLE status or deenergize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or~~

c. ~~b.~~ Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.4.~~2~~ 1 All containment penetration conductor overcurrent protective devices and feeder breakers to isolation transformers between 480 V Class 1E busses and non-Class 1E equipment ~~given in Table 3.8-1~~ shall be demonstrated OPERABLE:

a. At least once per 18 months:

1) By verifying that the ~~medium voltage [4-15~~ 13.8 kV] circuit breakers are OPERABLE by selecting, on a rotating basis, at least ~~10% of~~ one the circuit breakers of each voltage level, and performing the following:

a) A CHANNEL CALIBRATION of the associated protective relays,

b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed, and

Insert to Page 3/4 8-17

a. Restore the protective device or feeder breaker to OPERABLE status or deenergize the circuits by racking out or removing the inoperable circuit breaker or protective device and tripping the associated backup circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the inoperable circuit breaker or protective device racked out or removed at least once per 31 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices or feeder breakers in circuits which have their backup circuit breakers tripped, their inoperable circuit breaker racked out or removed, or

b. Deenergize the circuits by racking out or removing the inoperable circuit breaker or protective device within 72 hours, declare the affected system or component inoperable, and verify the inoperable circuit breaker or protective device racked out or removed at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices or feeder breakers in circuits which have their inoperable circuit breaker racked out, or removed, or

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ~~10%~~ one of ~~all~~ the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2) By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current ~~with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved~~. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested; ~~and~~

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to ensure that it is within the tolerances specified by the manufacturer.

TABLE 3.8-1

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	SYSTEM POWERED
1. 6900 VAC (Primary breaker) (Backup breaker)	Reactor Coolant pump 1 2 3 4
2. 480 VAC from MOAD Centers List all; primary breakers Backup breakers Backup breakers	
3. 480 VAC from MCC List all; primary breakers Backup breakers Backup breakers	
4. 125V DC Lighting List all; primary breakers Backup breakers Backup breakers	
5. 440 VAC CRDM Power Primary breakers Backup breakers Backup breakers	

ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

SAFETY-RELATED

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION ~~AND~~ BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

safety-related motor-operated

3.8.4.~~3~~ 2 The thermal overload protection ~~and~~ bypass devices, ~~integral with the motor starter~~ of each valve ~~listed in Table 3.8-2~~ shall be OPERABLE.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With ~~one or more of~~ the thermal overload protection ~~and/or~~ bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected valve(s).

for any one or more safety-related motor-operated valve

SURVEILLANCE REQUIREMENTS

~~4.8.4.3 The above required thermal overload protection and bypass devices shall be OPERABLE:~~

~~a. At least once per 18 months, by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST of the bypass circuitry for those thermal overload devices which are either:~~

~~1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or~~

~~2. Normally in force during plant operation and bypassed under accident conditions.~~

~~b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:~~

~~1. All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.~~

~~2. All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.~~

See insert to page 3/4 8-19.

INSERT TO PAGE 3/4 8-19

4.8.4.2 The above required thermal overload protection bypass devices shall be verified to be OPERABLE.

a. Following maintenance on the valve motor starter, and

b. Following any periodic testing during which the thermal overload device was temporarily placed in force.

c. At least once per 18 months, during shutdown.

TABLE 3.8-2

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (Continuous)(Accident Conditions)(No)	SYSTEM(S) AFFECTED

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.8

3.8.1.1.b:

The VEGP diesel engines are not equipped with engine-mounted fuel tanks.

The requirement for lubricating oil storage minimum volume and transfer capability was deleted on the basis that the VEGP diesel generator lube oil system is integral to the diesel package and will be covered under the definition of operability of the diesel itself. See figure 9.5.4-1 and response to Question 430.35 in the VEGP FSAR.

3.8.1.1, Action a:

This action statement was revised to clarify that, given the loss of an offsite circuit, the untested diesel generator may already be operating; in which case, testing would be unnecessary. In addition, the action statement was revised to specify Hot Standby in 6 hours and Cold Shutdown within the following 30 hours for the sake of consistency.

3.8.1.1, Action b:

This action statement was revised to clarify the point at which a diesel generator is considered to be inoperable. The requirement to be in Hot Shutdown within 12 hours and Cold Shutdown within the following 24 hours was revised to require Hot Standby in 6 hours and Cold Shutdown within the following 30 hours for the sake of consistency.

3.8.1.1, Action c:

This action statement was inserted to provide for the case where an offsite circuit and a diesel generator may be inoperable. As part of the equipment is restored, action is transferred to an earlier action statement. Explicit language has been included to highlight this transfer and clarify the starting time for the requirements of the action statement being entered. The transfer statement also clarifies that if a diesel generator has already been successfully tested as part of the present action statement, it does not have to be retested.

3.8.1.1, Action e:

This action statement has been revised to include explicit language which transfers action to a previous action statement when one offsite source has been restored and one remains inoperable. In addition, the requirement to perform surveillance on the diesel generators within 1 hour and once per 8

hours thereafter was revised to within 8 hours. This was done to reduce operating the diesels in parallel with a degraded system; reduce demands on the operators' time for testing; and reduce the number of tests required of the diesel generators. As presently worded, the Standard Technical Specifications could result in as many as nine tests of each diesel generator (a total of 18 tests for two diesels) over a 72-hour period. We believe this kind of testing to be excessive, especially in view of the fact that current industry experience indicates that excessive testing can be detrimental and could cause loss of the EDG when it is needed most. Further, since this type of testing involves operating the EDG in parallel with a degraded system, the possibility of losing the EDG during such testing is increased. The EDGs are designed and intended to be standby power sources. Therefore, the negative implications of such a test are not trivial.

There is also the consideration for operator actions during plant abnormalities. Operator capabilities to diagnose and track plant conditions have been increased and priority of operator actions has been given further attention. Actions to assure adequate reactor core cooling may continue over hours and have priority over the testing of redundant equipment. Operator actions that could distract the operator, require his time to be spent on lower priority actions, or have the potential to unnecessarily generate doubts or confusion, need to be minimized.

3.8.1.1, Action f:

The action statement has been revised to transfer action to a previous action statement as equipment is restored to operability. The transfer statement also clarifies that if a diesel generator has already been successfully tested as part of the present action statement, it does not have to be retested.

4.8.1.1.1.b:

This surveillance requirement is not applicable to VEGP on the basis that the Class 1E distribution system has two independent ac power trains, each fed from an independent Class 1E bus with immediate access to the offsite transmission network through a physically independent circuit. A failure of a single active component in the onsite electric distribution system will not prevent the safety-related systems from performing their function. The VEGP design of the offsite transmission network to the onsite safety-related loads, with power from the offsite transmission network for each circuit available immediately, provides the independence and redundancy required to effect a safe shutdown of the plant assuming a single failure. This is accomplished without taking credit for alternate feed capability.

4.8.1.1.2.a.1 &
4.8.1.1.2.a.3:

There is no engine-mounted fuel tank.

4.8.1.1.2.a.4 (STS):

See the justification provided for 3.8.1.1.b.

4.8.1.1.2.a.4:

This surveillance has been revised to require that the generator voltage and frequency be at acceptable levels within 11.4 seconds after the start signal. The specification on engine speed is redundant to the requirement on frequency and is therefore not necessary. From the standpoint of the safety-related loads, the proper voltage and frequency are the important parameters. This surveillance has also been revised by deleting the words "from ambient conditions" since each start for the purpose of this surveillance will be preceded by a prelube period in accordance with manufacturer's recommendations. The primary purpose of routine (monthly) testing should be to implement the recommended testing and to verify starting and load handling capability, rather than to simulate the design basis accident conditions.

4.8.1.1.2.a.5 and Footnote *:

This surveillance has been revised to eliminate the fast loading requirement on the basis that routine (monthly) testing should demonstrate starting and load handling capability rather than simulating design basis accident conditions. The fast loading will be done every 184 days as per new Surveillance 4.8.1.1.2.f. Also, a loading band has been specified rather than a target load. This has its basis in the fact that some operating plants have been forced to routinely overload their diesels during testing man effort to ensure that minimum test requirements are met. When instrument uncertainties are factored in, the overloading could be significant. The monthly test should exercise the diesel generator, confirm its operability, and detect degradation or a failure before a second diesel generator failure is likely to occur. During the 18-month testing, the test loads envelope the calculated accident loads. It should not be necessary or desirable to envelope the design basis accident loads, which might occur in 10,000 years, by a test that is repeated 12 times each year. The band specified (6100 - 7000 kW) is based on a maximum auto-connected load of 6032 kW and the continuous duty rating of 7000 kW. Footnote * has been revised since 4.8.1.1.2.a.5 no longer requires fast loading.

4.8.1.1.2.a.7:

This surveillance requirement was added to replace the requirement to start the diesels at least 5 times given a certain pressure in the air start receivers. Since the volume of the air start receivers is fixed and the capability to start the diesels at least 5 times without compressor assistance is demonstrated during preoperational testing, verifying that adequate pressure is available should be sufficient to demonstrate the capability to start the diesels. In addition, diesel generator start time is verified on at least monthly intervals. This revision reduces the required testing while maintaining the intent of the specification.

4.8.1.1.2.b:

There is no engine-mounted fuel tank.

4.8.1.1.2.c through 4.8.1.1.2.e:

The fuel oil sampling program has been revised based on a commitment to the NRC as stated in the response to Question 430.9 of the VEGP FSAR. This program is consistent with what has already been implemented at the McGuire Nuclear Station.

4.8.1.1.2.f:

See the justification provided for 4.8.1.1.2.a.5.

4.8.1.1.2.g, Footnote #:

This footnote was revised for clarification.

4.8.1.1.2.g.1:

The words "its manufacturer's" were deleted on the basis that GPC is committed to the TDI Owners Group Program rather than relying solely on the manufacturer's maintenance recommendations.

4.8.1.1.2.g.2:

The word "diesel" was inserted in the first line to clarify that this test is intended to verify that the engine governor is operating properly.

The value of 484 rpm is based on the TDI Owners Group Design Review and Quality Revalidation Report for VEGP. An analysis of engine operation indicated a need to limit engine speed to less than or equal to 484 rpm to avoid critical speed resonance at 496 rpm.

4.8.1.1.2.g.3:

Again, the word "diesel" was inserted for clarification.

4.8.1.1.2.g.6.c:

The words "upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal" were deleted on the basis that they are redundant to the first line of 4.8.1.1.2.g.6.

4.8.1.1.2.g.7:

This surveillance has been revised to reflect a band for loading based on the reasoning presented in the discussion of revisions to Surveillance Requirement 4.8.1.1.2.a.5. In this case, however, it is desirable to test as close to the 2-hour rating as possible without overloading the diesel. For this reason, a target value of 7650 kW has been proposed for the 2-hour portion of this test.

The words "within ( ) seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits" were deleted on the basis that there is no requirement to have a separate start to begin the 24-hour test. This test could be performed in conjunction with another test which has already started the diesel and required that start time, voltage, etc. be verified. The footnote ** was added since the diesel generator is paralleled with the grid. Therefore, disturbances attributable to the grid should not invalidate the 24-hour test.

The reference surveillance to be performed within 5 minutes of the 24-hour test was revised to 4.8.1.1.2.g.4.b on the basis that this test requires a higher loading for the diesel generator and has less impact on the plant than repeating the ESF actuation test signal and loss of offsite power start test. The intent of the surveillance remains unchanged, ie., the diesel is capable of emergency start after the 24-hour test.

4.8.1.1.2.g.8:

The 2000-hour rating for the VEGP diesels is 7000 kW. However, the maximum auto-connected load is 6032 kW. Therefore, a test value of 6100 kW will ensure that the auto-connected loads are within design limits.

4.8.1.1.2.g.11:

There is no engine-mounted fuel tank.

4.8.1.1.2.g.13 (STS):

The VEGP design is such that the diesel is prevented from starting any time the turning gear (barring device) is engaged or the emergency stop is actuated.

4.8.1.1.2.g.14:

This item was deleted as previously discussed under 4.8.1.1.2.a.7.

4.8.1.1.2.h (STS):

This surveillance requirement was deleted on the basis that starting both diesel generators simultaneously is not a test designed to demonstrate freedom from interdependencies. Design controls are in place to prevent interdependencies at the design and construction stage and controls will continue to be enforced during the operations phase to preclude modifications which could result in interdependencies.

4.8.1.1.3:

This reporting requirement has been revised to an annual diesel generator reliability data report. The information proposed for this report is based on Generic Letter 84-15 and is contained in Section 6.9 of the Technical Specifications. This revision is also consistent with the intent of Generic Letter 83-43.

Table 4.8-1:

The proposed test schedule bases accelerated test requirements only on the number of failures in the last 20 valid tests. Basing the test frequency on failures during the previous 100 valid tests is deleted because it is not an accurate measure of the reliability of the diesel. Nearly 2 years are required to complete 100 tests at the accelerated test frequency. Thus, an isolated problem which caused several test failures before being corrected could result in accelerated testing for a 2-year period. This could be excessive testing which is counter to vendor and industry recommendations for optimizing diesel generator reliability. The proposed test schedule is consistent with the intent of Generic Letter 84-15. In addition, footnote * was revised to be consistent with the fact that the 184-day testing requirements are listed under 4.8.1.1.2.f in our markup.

3.8.1.2.b:

There is no engine-mounted fuel tank.

The action statement was revised to reference Specification 3.4.9.3 for pressure relief capability for the RCS. The words "depressurize and vent" were deleted to allow credit for the RHR suction relief valves. The RHR suction isolation valves can be manually opened in the event of a loss of power.

4.8.1.2:

Surveillance Requirement 4.8.1.1.3 was deleted to be consistent with the fact that 4.8.1.1.3 was deleted from Specification 3/4.8.1.1. See the justification provided for the deletion of 4.8.1.1.3.

3.8.2.1 a & b and Action Statements:

This limiting condition for operation and the action statements were revised to reflect the plant-specific design for the 125 V-dc system. See subsection 8.3.2 of the VEGP FSAR for a system description.

4.8.2.1.b.2 & 4.8.2.1.c.3:

The limit on connection resistance was revised to "20 percent over the average as measured during the plant preoperational tests" on the basis that this is consistent with the recommendations of IEEE Standard 450.

4.8.2.1.C.4:

This surveillance has been revised to reflect the VEGP design.

Table 4.8-2:

The limit on float voltage was revised from 2.07 V to 2.10 V on vendor recommendations. The specific gravity was revised to reflect the VEGP specific manufacturer's full charge specific gravity of 1.210.

3/4.8.2.2:

This specification has been revised so that it is applicable to the VEGP specific design for the 125 V-dc system. See subsection 8.3.2 of the VEGP FSAR for a system description.

The action statement was revised to reference Specification 3.4.9.3 for pressure relief capability for the RCS as discussed in the justification for 3.8.1.2.b.

3/4.8.3.1 and 3/4.8.3.2:

These specifications were revised to reflect the VEGP specific design in as clear and straightforward a manner as possible. See section 8.3 of the VEGP FSAR for a system description.

The action statement of 3/4.8.3.2 was revised to reference Specification 3.4.9.3 for pressure relief capability for the RCS as discussed in the justification for 3.8.1.2.b.

3/4.8.4.1 (STS):

This section is not applicable to VEGP based on the fact that all the containment electrical penetration and penetration conductors will be protected by demonstrating the operability of primary and backup overcurrent protection circuit breakers during periodic surveillance, as required by Specification 3/4.8.4.1.

3.8.4.1:

This specification has been revised to include-the feeder breakers to the isolation transformers between 480 V Class 1E busses and non-Class 1E equipment as discussed in Section 8.4.5 of the VEGP SER.

Table 3.8-1 was deleted on the basis that such a lengthy listing does not provide useful information to the operator in ensuring Technical Specification compliance. This listing can be maintained in plant procedures without affecting the intent of the Technical Specifications.

Action Statement a. has been revised to reflect the fact that the inoperable circuit breaker or protective device will be racked out in addition to tripping the backup breaker. Therefore, a 31-day surveillance interval should be sufficient to ensure that the affected circuit(s) remain deenergized until the protective device(s) are restored to operable status. This is consistent with other action statements or LCO's which require that equipment be rendered inoperable during certain phases of plant operation (reference Surveillance Requirement 4.5.3.2, for example).

Action Statement b has been added to reflect the STS action of 7-day surveillance for those circuit breakers or protective devices where it may not be desirable to trip the backup breaker in addition to racking out the circuit breaker or protective device. For example, if the backup circuit breaker feeds more than one circuit breaker or protective device, it would not be desirable to deenergize all the circuits associated with the backup breaker.

The above revisions provide additional flexibility without altering the intent of the specification.

4.8.4.1:

There are only four circuit breakers which fall into the category of 4-15 kV breakers. These are the 13.8 kV breakers for the reactor coolant pumps. Therefore, 4.8.4.2.a has been revised to reflect this. See figure 8.3.1-1 sheet 1, of the VEGP FSAR. In addition, the feeder breakers to isolation transformers between 480 V Class 1E busses and non-Class 1E equipment have been included as discussed in Section 8.4.5 of the VEGP SER. Surveillance Requirement 4.8.4.2.a.2 has been revised to be less prescriptive on the basis that different breakers are tested by different methods. Therefore, such a prescriptive statement as contained in the Standard Technical Specifications is not practical for all low voltage circuit breakers. In general, the lower voltage circuit breakers will be tested in a more conservative fashion than called for in the Standard Technical Specifications.

3/4.8.4.2:

This specification was revised to reflect the fact that prior to core loading and during plant operation, the thermal overload relay trip contacts for all of the Class 1E valves are permanently bypassed with jumpers, in accordance with Regulatory Guide 1.106. See paragraph 8.3.1.1.2.k.5 of the VEGP FSAR.

Table 3.8-2 was deleted on the basis that this listing does not provide information which is essential for the operator in ensuring compliance with Technical Specifications. This list can be maintained in procedures without affecting the intent of the specification.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to ~~[~~2000~~]~~ ppm.

APPLICABILITY: MODE 6.~~*~~

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to [2000] ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

~~4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:~~

~~a. Removing or unbolting the reactor vessel head, and~~

~~b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.~~

4.9.1.~~2~~ 1 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.~~3~~ 2 Valves 1208-U4-175, 1208-U4-177, 1207-U4-183, and 1208-U4-176 ~~[Isolation of unborated water sources]~~ shall be verified closed and secured in position by mechanical stops ~~or by removal of air or electrical power~~ at least once per 31 days.

~~*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.~~

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1) Closed by an isolation valve, blind flange, or manual valve, or

2) Be capable of being closed by an OPERABLE automatic containment ~~purge and exhaust~~ ventilation isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment ~~purge and exhaust~~ ventilation isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment ~~purge and exhaust~~ ventilation isolation valves per the applicable portions of Specification 4.6.~~4~~ 3.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

1) A minimum capacity of [2750] pounds, and

2) An overload cutoff limit less than or equal to [2700] pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1) A minimum capacity of [610] pounds, and

2) A load indicator which shall be used to prevent lifting loads in excess of [600] pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least [2750] pounds and demonstrating an automatic load cutoff when the crane load exceeds [2700] pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least [610] pounds.

3/4.9.6 REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The refueling machine used for movement of fuel assemblies having:

1. A design rated load of 3166 pounds on the hoist (1966 pounds on the gripper).
2. Automatic cutoffs with the following setpoints relative to the suspended weight of the assembly in water:
   a. Primary overload is:
      1. Plus 250 pounds for wet conditions.
      2. Plus 350 pounds for dry conditions.
   b. Secondary overload is:
      1. Plus 150 pounds above primary overload.
   c. Load reduction is:
      1. Minus 250 pounds for both wet and dry conditions.

b. The auxiliary hoist used for latching and unlatching drive rods having:

1. A minimum capacity of 3000 pounds, and
2. A 1000 pound (minimum) load indicator which will be used to monitor lifting loads for these operations.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for the refueling machine and/or auxiliary hoist OPERABILITY not satisfied, suspend use of any inoperable refueling machine and/or auxiliary hoist from operations involving the movement of fuel assemblies and/or drive rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each refueling machine used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to start of such operations by performing a load test of at least 125% of the design rated load and by demonstrating an automatic load cutoff when the refueling machine load exceeds the setpoints of Specification 3.9.6.a.2.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 1250 pounds.

REFUELING OPERATIONS

3/4 9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of (Later) pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With irradiated fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of (Later) pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) ~~loop~~ train shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR ~~loop~~ train OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR ~~loop~~ to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

train

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least one RHR ~~loop~~ train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to [2800] gpm at least once per 12 hours.

*The RHR ~~loop~~ train may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) ~~loops~~ trains shall be OPERABLE, and at least one RHR ~~loop~~ train shall be in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR ~~loops~~ trains OPERABLE, immediately initiate corrective action to return the required RHR loops trains to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR ~~loop~~ train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR ~~loop~~ train to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR ~~loop~~ train shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to ~~[2800]~~ (Later) gpm at least once per 12 hours.

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*Prior to initial criticality, the RHR ~~loop~~ train may be removed from operation for up to 1 hour per ~~8~~ 2 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

3/4.9.9 CONTAINMENT VENTILATION ~~PURGE AND EXHAUST~~ ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation ~~Purge and Exhaust~~ Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation ~~Purge and Exhaust~~ Isolation System inoperable, close each of the ventilation ~~purge and exhaust~~ penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation ~~Purge and Exhaust~~ Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment ~~purge and exhaust~~ isolation occurs ~~on manual initiation and~~ on a High Radiation test signal from each of the containment radiation monitoring instrumentation channels. Ventilation

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

FUEL ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.9.10.1 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies within the containment when the fuel assemblies being moved are irradiated.

ACTION

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.1 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies.

WATER LEVEL- REACTOR VESSEL

CONTROL RODS

LIMITING CONDITION FOR OPERATION

3.9.10.2 At least 23 feet of water shall be maintained over the top of the irradiated fuel assemblies within the reactor pressure vessel.

APPLICABILITY: During movement of control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirement of the above specification not satisfied, suspend all operations involving movement of control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10.2 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of control rods within the reactor vessel.

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations ~~with loads in the fuel storage areas~~ over the spent fuel pool and restore the water level to within its limit within 4 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

3/4.9.12 FUEL STORAGE POOL AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 Two independent Fuel Storage Pool Air Cleanup Systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one Fuel Storage Pool Air Cleanup System inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE Fuel Storage Pool Air Cleanup System is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no Fuel Storage Pool Air Cleanup System OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one Fuel Storage Pool Air Cleanup System is restored to OPERABLE status.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12 The above required Fuel Storage Pool Air Cleanup Systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ____ cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

3) Verifying a system flow rate of ____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%.

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ____ cfm ± 10%,

2) Verifying that on a High Radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks,

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to [1/4] inch Water Gauge relative to the outside atmosphere during system operation,

4) Verifying that the filter cooling bypass valves can be manually opened, and

5) Verifying that the heaters dissipate _____ ± _____ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _____ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% of less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.9

3/4.9.1:

Footnote * was deleted as redundant since it mer ely repeats the definition for Mode 6

Surveillance Requirement 4.9.1.1 was deleted since this requirement is provided within the definition of Mode 6 which must be met prior to entry into Mode 6 as required by Specification 3.0.4.

This is consistent with the WOG "Short-Term Tech. Spec. Improvements" which were prioritized as a "Staff Action" item during the January 29, 1986 meeting between the WOG Tech. Spec. Subcommittee and the NRC Staff.

4.9.1.2:

Reference to air and electrical power was deleted since the valves in question are manual valves.

3/4.9.4 & 3/4.9.9:

"Purge and Exhaust" was replaced with "Ventilation" to reflect plant-specific terminology. See subsection 6.2.4 of the VEGP FSAR.

3/4.9.6:

The Generic STS Specification 3/4.9.6 (Manipulator Crane) was deleted in its entirety. A new Specification 3/4.9.6 (Refueling Machine) was inserted in its place. The new specification reflects the installed equipment in Plant Vogtle. See subsection 9.1.4 of the VEGP FSAR.

3.9.7:

The addition of the word "irradiated" reflects the intent of this specification.

3.9.8.1:

The change reflects the VEGP specific terminology "Residual Heat Removal Train" instead of "Residual Heat Removal Loop." See subsection 5.4.7 of the VEGP FSAR.

3.9.8.2:

The change reflects the VEGP specific terminology "Residual Heat Removal Train" instead of "Residual Heat Removal Loop." See subsection 5.4.7 of the VEGP FSAR.

This revision has been approved by the NRC Staff at a recent operating plant and is consistent with the WOG short-term recommendations as discussed in the Jauuary 29, 1986 meeting between the WOG Tech. Spec. Subcommittee and the NRC Staff. This item was prioritized as a "Staff Action" item at that meeting.

The revision of footnote * allowing the RHR train to be out of operation for 1 hour out of every 2 hours (initial core load) was made on the basis that, since there is no decay heat involved, there should be no problem with boron stratification or precipitation. Also, there is no likelihood of a boron dilution transient as discussed in 15.4.6 of the VEGP FSAR.

3/4.9.9:

This specification was revised to reflect VEGP specific terminology and the fact that the design does not include a specific hand-switch for manual initiation. See subsection 6.2.4 of the VEGP FSAR.

3/4.9.10:

This specification was split into two specifications to facilitate the connection of the drive rods to the control rods. When fuel assemblies are being moved, at least 23 feet of water is required above the vessel flange to provide the necessary shielding. However, when fuel assemblies are not being moved but the drive rods are being connected to the control rods, 23 feet of water above the vessel flange is not necessary for proper shielding. By requiring 23 feet of water above the top of the irradiated fuel assemblies, the necessary shielding is maintained, and the operation of connecting the drive rods to the control rods is facilitated. This has been approved by the NRC and implemented in the Farley Nuclear Plant Technical Specifications.

3/4.9.11:

The replacement of the phrase "with loads in the fuel storage areas" with "over the spent fuel pool" in this specification's action statement provides a more defined action and improves the consistency between the action and the assumptions used in the accident analysis.

3/4.9.12:

The VEGP is equipped with a fuel handling building post accident exhaust system. However, Chapter 15 of the SER specifically states that no credit was assumed for the operation of these filters in mitigating the consequences of a fuel handling accident in the fuel handling building. Consequently, this specification has been deleted since it is not required to validate an assumption made in the safety analysis.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any ~~full-length~~ control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all ~~full-length~~ control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each ~~full-length~~ control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each ~~full-length~~ control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within ~~24 hours~~ prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

7 days

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.3.3, and

b. Specification 4.2.3.2.

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set ~~at less than or equal to 25% of RATED~~ in accordance with Table 2.2-1 ~~THERMAL POWER~~, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to ~~[531]~~ 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than ~~[531]~~ 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to ~~[531]~~ 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of STARTUP and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during STARTUP and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating STARTUP and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of the following requirements may be suspended:

a. Specification 3.4.1.1 - During the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

   1) The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

   2) The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set in accordance with Table 2.2-1.

b. Specification 3.4.1.2 - During the performance of hot rod drop time measurements in MODE 3 provided at least two reactor coolant loops as listed in Specification 3.4.1.2 are OPERABLE.

APPLICABILITY: During operation below the P-7 Interlock Setpoint or performance of hot rod drop time measurements.

ACTION:

a. With the THERMAL POWER greater than the P-7 Interlock Setpoint during the performance of startup and PHYSICS TESTS, immediately open the Reactor trip breakers.

b. With less than the above required reactor coolant loops OPERABLE during performance of hot rod drop time measurements, immediately open the reactor trip breakers and comply with the provisions of the ACTION statements of Specification 3.4.1.2.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

4.10.4.3 At least the above required reactor coolant loops shall be determined OPERABLE within 4 hours prior to initiation of the hot rod drop time measurements and at least once per 4 hours during the hot rod drop time measurements by verifying correct breaker alignments and indicated power availability and by verifying secondary side wide range water level to be greater than or equal to 17%.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual ~~full-length~~ shutdown and control rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4, and 5 during performance of rod drop time measurements.

ACTION:

With the Position Indication Systems inoperable or with more than one bank of rods withdrawn, immediately open the Reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required Position Indication Systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the Demand Position Indication System and the Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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*This requirement is not applicable during the initial calibration of the Digital Rod Position Indication System provided: (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.10

3/4.10:

Since the plant has only full-length control rods, the words "full-length" have been deleted as redundant. See paragraph 4.2.2.3 of the VEGP FSAR.

4.10.1.2:

The purpose of Surveillance Requirement 4.10.1.2 is to assure the reliability of the reactor control rod insertion capability prior to reducing shutdown margin below specified levels during low power (less than 5 percent) physics tests. The revision from 24 hours to 7 days will allow a more expeditious startup following initial criticality and subsequent refuelings without significantly increasing the probability that a stuck rod will occur in conjunction with a positive reactivity addition transient. This has been implemented at San Onofre.

3.10.3:

LCO item b was deemed redundant to Table 2.2-1 (which sets both the IR and PR low setpoint at ≤ 25 percent RTP) and was modified accordingly.

3.10.4:

LCO item a.2 was deemed redundant to Table 2.2-1 (which sets both the IR and PR low setpoint at ≤ 25 percent RTP) and was modified accordingly.

Special Test Exception 3.10.4(b) allows the suspension of 3.4.1.2 during the no flow hot rod drop time test in Mode 3 provided at least two reactor coolant loops are OPERABLE. The two reactor coolant loops would provide redundant decay heat removal paths. An uncontrolled RCCA bank withdrawal during performance of the subject testing in Mode 3 is not anticipated since the operator would be actively monitoring the rod control system and would be able to detect and correct any problems arising with the rod control system. This special test exception has already been approved and implemented in the Byron Unit 1 Technical Specifications.

4.10.4.3:

This surveillance requirement was added to provide a means to determine OPERABILITY of the two reactor coolant loops.

3.10.5:

The words "full-length" are unnecessary since the VEGP does not use part-length rods. See paragraph 4.2.2.3 of the VEGP FSAR.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-~~3~~1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (µCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	5×10^{-7}
a. Waste Monitor Tank 1901-T6-009			I-131	1×10^{-6}
b. Waste Monitor Tank 1901-T6-010	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
c. Drainage of Systems	P Each Batch	M Composite(4)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite(4)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5)	Continuous(6)	W Composite(6)	Principal Gamma Emitters(3)	5×10^{-7}
a. Waste water retention basin			I-131	1×10^{-6}
~~b.~~	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
~~c.~~	Continuous(6)	M Composite(6)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous(6)	Q Composite(6)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

(2)A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed by a method described in the ODCM to assure representative sampling.

TABLE 4.11-1 (Continued)

TABLE NOTATIONS (Continued)

(3)The principal gamma emmiters for which the LLD specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of 5×10^{-6}. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(4)A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5)A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6)To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-~~3~~ 1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. ~~This Special Report shall also include: (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR Part 141, Safe Drinking Water Act.*~~

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

~~*The requirements of ACTION a.(1) and (2) are applicable only if drinking water supply is taken from the receiving water body within 3 miles of the plant discharge. In the case of river-sited plants this is 3 miles downstream only.~~

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-~~3~~) would exceed ~~0.06~~ 0.18 mrem to the whole body or 0.2 mrem to any organ ~~in a 31-day period~~ per calendar quarter.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.~~1~~ Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

~~4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.~~

RADIOACTIVE EFFLUENTS

LIQUID HOLDUP TANKS*

LIMITING CONDITION FOR OPERATION

3.11.1.4 The quantity of radioactive material contained in each outside temporary ~~of the following unprotected outdoor~~ tank~~s~~ shall be limited to less than or equal to 10 Curies, excluding tritium and dissolved or entrained noble gases:

~~a.~~ ________

~~b.~~ ________

~~c.~~ ________

~~d. Outside temporary tank~~

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of either the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4, or provide notification to the Commission pursuant to Specification 6.9.2 within 30 days, in lieu of any other report

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of either radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank, or each batch of radioactive material prior to its addition to the tank.

____

*Tanks included in this specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5.1-~~3~~ 1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined semi-annually to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

space

TABLE 4.11-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/ml)
1. Waste Gas Storage Tank Decay	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge ~~or Vent~~ 24" or 14"	P Each PURGE(3) Grab Sample	P Each PURGE(3)	Principal Gamma Emitters(2)	1×10^{-4}
		M	H-3 (oxide)	1×10^{-6}
3. a. Plant Vent	M(3),(4),(5) Grab Sample	M(3)	Principal Gamma Emitters(2)	1×10^{-4}
			H-3 (oxide)	1×10^{-6}
b. ~~Fuel Storage Area Ventilation~~ Condenser Air Ejector & Steam Packing Exhaust	M(5)(6) Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
			H-3 (oxide)	1×10^{-6}
c. Auxiliary Bldg, Radwaste Area, ~~SGB Vent,~~ Solidification Building Others	M Grab Sample	M	Principal Gamma Emitters(2)	1×10^{-4}
4. All Release Types as listed in ~~1., 2.,~~ and 3.1 and 3.3 above. Also 3.b when a primary to secondary leak has been confirmed.	Continuous(6)	W(7) Charcoal Sample	I-131	1×10^{-12}
	Continuous(6)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(6)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(6)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}

(1902-V6-001, 1906-V6-002, 1902-V6-003, 1906-V6-004, 1902-V6-005, 1906-V6-006, 1902-V6-007, 1906-V6-009, A-1902-V6-010)

TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate. (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification applies include the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas releases and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in Iodine and particulate releases. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 in the format outlined in Regulatory Guide 1.21, Appendix B, Revision 1, June 1974.

(3) Sampling and analysis shall also be performed following shutdown, startup, or a continuous increase or decrease in THERMAL POWER ~~change~~ exceeding 15% of RATED THERMAL POWER within a 1-hour period.

(4) Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

(5) Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

(6) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(7) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or continuous increase or decrease in THERMAL POWER ~~change~~ exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the reactor coolant has not increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

(8) Sampling not required unless a primary-to-secondary leak is confirmed.

This requirement does not apply if (1) analysis shows that the Dose Equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3; (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3; and (3) a purge is in progress.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-~~3~~ 1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-~~3~~1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the GASEOUS WASTE PROCESSING ~~WASTE GAS HOLDUP~~ SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-~~3~~ 1) would exceed either:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.~~1~~ Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

~~4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and WASTE GAS HOLDUP SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.~~

EXPLOSIVE GAS MIXTURE ~~[Systems not designed to withstand a hydrogen explosion]~~

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE [Systems designed to withstand a hydrogen explosion]

LIMITNG CONDITION FOR OPERATION

3.11.2.5 The concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM exceeding the limit, restore the concentration to within the limit within 48 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the WASTE GAS HOLDUP SYSTEM shall be determined to be within the above limits by continuously monitoring the waste gases in the WASTE GAS HOLDUP SYSTEM with the hydrogen or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.11.

RADIOACTIVE EFFLUENTS

GAS ~~STORAGE~~ DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas ~~storage~~ decay tank shall be limited to less than or equal to 2.0 x 10^5 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas ~~storage~~ decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6.1 The quantity of radioactive material contained in each gas ~~storage~~ decay tank shall be determined to be within the above limit at least once per ~~24~~ 7 days ~~hours~~ when radioactive materials ~~are being~~ have been added to the tank, during the previous 7 days.

4.11.2.6.2 In the event of confirmed major fuel failure (>1%) the quantity of radioactive material contained in each waste gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials have been added to the tank in the previous 24 hours.

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive wastes shall be solidified or dewatered in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.3 SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM;

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units (including outside storage tanks etc.) to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units (including outside storage tanks etc.) shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.11

3/4.11:

Instrument tag numbers were added as appropriate to assist the operator in ensuring compliance with these Technical Specifications.

3.11.1.1, Action b:

10 CFR 20.106(a) states "For purposes of this section concentrations may be averaged over a period not greater than one year." Therefore, it is not reasonable to require shutdown of the reactor within 1 hour when 10 CFR 20, Appendix B, Table II, Column 2 concentrations are exceeded.

In addition, requiring shutdown of the reactor within 1 hour when Appendix B concentrations are exceeded is not consistent with 10 CFR 50.73 (B) (viii) B which allows levels of twice Appendix B values when averaged over 1 hour as a criterion for determining whether the event is reportable.

Further, releases from radwaste systems may not be related to the operational mode of the reactor; therefore, reactor shutdown may not be appropriate as a means of restoring concentrations to within Appendix B limits. Therefore, the provisions of Specifications 3.0.3 and 3.0.4 should not apply.

3.11.1.2, Action a:

The deletion of the additional special report information is justified on the basis that there is no drinking water taken from the river within 3 miles of plant discharge as noted by footnote *.

3.11.1.3:

The change that revises the dose projection from a 31-day period to a quarterly basis is for operational convenience and does not alter the intent of the specification in any way since the corresponding dose limit has been adjusted for a quarterly projection.

4.11.1.2 (STS):

This surveillance requirement was deleted on the basis that is redundant to the requirements of 3.11.1.1 and 3.11.1.2.

3.11.1.4:

The outdoor tanks which contain radioactive material at VEGP are Seismic Category I and are missile protected. Consequently, this specification should apply only to temporary tanks which are brought onsite for additional storage capacity. See paragraph 2.4.13.2 of the VEGP FSAR.

The referenced 10-curie limit is justified based on NUREG-0133.

3.11.1.4, Action a:

Action Statement a was modified to permit greater than 48 hours for reduction of tank contents so long as immediate action is taken (i.e., suspend additions to the tank) and an increased level of reporting is complied with. This would allow a more realistic time frame for the implementation of actions necessary tq reduce the tank contents to within the limit. For a temporary outside tank, this could involve delivery of an additional tank or modification of the existing tank to drain or pump the contents.

4.11.1.4:

The revision recognizes that a temporary tank may not allow a representative sample to be obtained. Analyzing each batch prior to addition to the tank will be the same as analyzing the total tank contents. Thus, the intent of the specification is preserved.

3.11.2.1, Action b:

According to Technical Specification BASES the annual dose limits are the doses associated with the concentrations of 10 CFR 20, Appendix B, Table II, Column 1. 10 CFR 20.106 (a) states "For purposes of this section concentrations may be averaged over a period not greater than one year." Therefore, it is not reasonable to require shutdown of the reactor within 1 hour when the dose rate(s) associated with Appendix B is(are) exceeded.

In addition, requiring shutdown of the reactor within 1 hour when a dose rate associated with Appendix B concentrations is exceeded is not consistent with 10 CFR 50.73 (b) (viii) A which allows levels of twice Appendix B values when averaged over 1 hour as a criterion for determining whether the event is reportable.

Further, releases from radwaste systems may not be related to the operational mode of the reactor; therefore, reactor shutdown may not be appropriate as a means of restoring concentrations to within Appendix B limits. Therefore, the provisions of Specifications 3.0.3 and 3.0.4 should not apply.

4.11.2.1.1:

The word "semiannually" was inserted on the basis that the determination will be made semiannually for the semiannual report.

Table 4.11-2:

See section 11.3 of the VEGP FSAR.

Item 1:

"Decay" was added to reflect plant-specific nomenclature.

Item 2:

"Vent" was deleted to reflect plant-specific design. Footnote 3 was revised to incorporate the same conditions used in footnote f on the basis that both footnotes are applied to sampling for principal gamma emitters. Footnote 3 was also revised to require sampling only when a purge is in progress.

Item 3a:

Footnote 5 was added since the fuel storage area ventilation exhausts through the plant vent.

Item 3b:

"Condenser Air Ejector and Steam Packing Exhauster" reflect plant-specific design. "Fuel Storage Area Ventilation" exhausts through the plant vent. Footnote 8 was added to clarify that sampling need not be performed unless a primary-to-secondary leak exists. See the justification provided for item 4 below.

Item 3c:

"Auxiliary Building" exhausts through the plant vent. The "Radwaste Solidification Building" identifies a plant-specific feature. The steam generator blowdown vent exhaust through the condenser air ejector and steam packing exhaust and was therefore deleted.

Item 4:

Types 1 and 2 (decay tanks and containment purge) need not be included since both are released via the plant vent and sampling at the plant vent is sufficient. Gaseous release type 3b need not be included unless a primary-to-secondary leak has been confirmed on the basis that unless such a leak exists the condenser air ejector and steam packing exhauster does not represent a release pathway. Specification 3/4.7.1.4 will ensure that primary-to-secondary leaks are identified in a timely manner via a gross radioactivity determination at least once per 72 hours.

Footnotes 3 and 7 were revised by the addition of the words "continuous increase or decrease in" for clarification. Without the revision, there was concern that a 15-percent change in power in a 1-hour period could be interpreted as, for example, the sum of a 5-percent decrease, a subsequent 5-percent increase, and another 5-percent decrease.

3.11.2.4:

The system terminology change from waste gas holdup system to gaseous waste processing system reflects plant-specific nomenclature. See section 11.3 of the VEGP FSAR.

4.11.2.4.2 (STS):

This surveillance requirement was deleted on the basis that it is redundant to the requirements of 3.11.2.1, 3.11.2.2, and 3.11.2.3.

3.11.2.5, Action Statements a and b, and 4.11.2.5:

The change in terminology (from waste gas holdup system to gaseous waste processing system) was made to reflect plant-specific nomenclature. See section 11.3 of the VEGP FSAR.

3/4.11.2.5, Explosive Gas Mixture (Systems designed to withstand a hydrogen explosion):

This specification was deleted since VEGP is not equipped with a gaseous waste processing system designed to withstand a hydrogen explosion.

3.11.2.6:

The change in terminology (from "Storage" to "Decay") reflects plant-specific nomenclature. See section 11.3 of the VEGP FSAR.

The basis for the 2.0 x 10^5 Ci for Xe-133 is outlined in NUREG-0133 and complies with the criteria in Branch Technical Position ESTB 11-5 as outlined in NUREG-0800.

4.11.2.6.1 & 4.11.2.6.2:

The proposed revision relaxes the surveillance requirements for quantifying the gas decay tank inventory from once per 24 hours when adding radioactive material to the tank to once per 7 days when activity has been added during the previous 7 days - in the absence of confirmed major fuel failure (> 1 percent).

a) The total noble gas inventory associated with 1-percent defects (i.e., approximately 90,000 Ci) is significantly below the activity inventory which could result in the total body exposure limit of 0.5 rem (300,000 Ci). Further, plant operation with fuel failures approaching or exceeding the equivalent of 1-percent fuel defects is not likely based on previous plant experience and other Technical Specification limitations (i.e., iodine RCS activity).

b) Based on the above, a daily sampling and analysis schedule is considered to be too frequent and inconsistent with the ALARA philosophy; i.e., it does not minimize occupational radiation exposure to the operators who obtain and analyze the samples.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program[(a)] shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental ~~Operating~~ Surveillance Report required by Specification 6.9.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the confirmed[(b)] level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter or after confirmation, whichever is later, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose* to a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental ~~Operating~~ Surveillance Report required by Specification 6.9.1.3.

*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With adequate samples of milk or ~~fresh leafy vegetation samples~~ vegetation permanently or persistently unavailable from one or more of the sample locations required by Table 3.12-1, ~~identify specific~~ efforts shall be made to find locations for obtaining replacement samples and ~~add them~~ if successful, these shall be added within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may ~~then~~ be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s), if any, with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s), if any, for obtaining samples, or the unavailability of suitable new locations.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

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a. The requirements for radiological environmental monitoring are the same for both units at the site. Thus, a single program including monitoring, land use survey, and quality assurance serves both units.

b. Defined as a confirmatory reanalysis of the original, a duplicate, or a new sample, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM*

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(2)	Thirty-six ~~Forty~~ routine monitoring stations ~~(DR1-DR40)~~ either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each meteorological sector in the general area of the SITE BOUNDARY (DR1-DR16); An outer ring of stations, one in each meteorological sector in approximately the 4 to 5 miles ~~6- to 8-km~~ range from the site (DR17-DR32); and The balance of the stations ~~(DR33-DR40)~~ to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.	Quarterly.	Gamma dose quarterly.

*The number, media, frequency, and location of samples ~~may vary from site to site. This table presents an acceptable minimum program for a site at which each entry is applicable. Local site characteristics must be examined to determine if pathways not covered by this table may significantly contribute to an~~ individual's dose and should be included ~~in the sample program. The code letters in parentheses, e.g.~~, DR1, A1~~, provide one way of defining sample locations in this specification that can be used to identify~~ the specific locations in the map(s) and ~~table in the ODCM~~.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
2. Airborne			
Radioiodine and Particulates	Samples from five locations ~~(A1-A5)~~: Three samples ~~(A1-A3)~~ from locations close to the ~~three~~ SITE BOUNDARY locations, in different sectors, ~~of the highest calculated annual average ground-level D/Q~~; One sample ~~(A4)~~ from the vicinity of a community having the highest calculated annual average ground-level D/Q; and One sample ~~(A5)~~ from a control location, as ~~for example 15 to 30 km distant and in the least prevalent wind direction~~ at about 10 miles distant or beyond.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly. Particulate Sampler: Gross beta radioactivity analysis following filter change;[3] and gamma isotopic analysis[4] of composite (by location) quarterly.
3. Waterborne			
a. Surface[5]	One sample upstream ~~(Wa1)~~. One sample downstream ~~(Wa2)~~.	Composite sample over 1-month period.[6]	Gamma isotopic analysis[4] monthly. Composite for tritium analysis quarterly.
~~b. Ground~~	~~Samples from one or two sources (Wb1, Wb2), only if likely to be affected[7]~~	~~Quarterly.~~	~~Gamma isotopic[4] and tritium analysis quarterly.~~

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[1]	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (Continued)			
b. ~~c.~~ Drinking	Two ~~One~~ samples at ~~of~~ each of one to three (Wc1 - Wc3) of the nearest water treatment plants ~~supplies~~ that could be affected by its discharge. Two ~~One~~ samples at ~~from~~ a control location (Wc4).	Composite sample of river water near intake at each water treatment plant over 2-week period[6] when I-131 analysis is performed; monthly composite otherwise, and grab sample of finished water at each water treatment plant every 2 weeks or monthly, as appropriate.	I-131 analysis on each sample ~~composite~~ when the dose calculated for the consumption of the water is greater than 1 mrem per year ~~[8]~~ [7]. Composite for gross beta and gamma isotopic analyses[4] monthly. Composite for tritium analysis quarterly.
c. ~~d.~~ Sediment from Shoreline	One sample from downstream area with existing or potential recreational value (Wd1).	Semiannually.	Gamma isotopic analysis[4] semiannually.
4. Ingestion			
a. Milk	Samples from milking animals[8] in three locations ~~(Ia1 - Ia3)~~ within ~~5 km~~ ≤ 3 mile distance having the highest dose potential. If there are none, then one sample from milking animals in each of three areas ~~(Ia1 - Ia3)~~ between ~~5 to 8 km~~ 3 to about 5 miles distant where doses are calculated to be greater than 1 mrem per yr. ~~[8]~~ [7] One sample from milking animals[8] at a control location ~~(Ia4)~~, about 10 miles distant or beyond and preferably in a ~~15 to 30 km distant and in the least prevalent~~ wind direction of lesser prevalence.	Semimonthly ~~when animals are on pasture; monthly at other times~~.	Gamma isotopic[4] ~~and I-131~~ analysis[9] semimonthly ~~when animals are on pasture; monthly at other times~~.

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS(1)	SAMPLING AND COLLECTION FREQUENCY	TYPE AND FREQUENCY OF ANALYSIS
4. Ingestion (Continued)			
b. Fish ~~and Inverte-brates~~	At least ~~One~~ one sample of ~~each~~ any commercially ~~and~~ or recreationally important species in vicinity of plant discharge area. ~~(Ib1 - Ib_)~~.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis(4) on edible portions.
At least one sample of any anadromous species in vicinity of plant discharge	At least ~~One~~ one sample of ~~same~~ species in areas not influenced by plant discharge ~~(Ib10 - Ib_)~~. Any commercially or recreationally important		
c. ~~Food Products~~ Grass or Leafy Vegetation	~~One sample of each principal class of food products from any area that is irrigated by water in which liquid plant wastes have been discharged (Ic1 - Ic_).~~	~~At time of harvest(9).~~ During spring spawning season.	~~Gamma isotopic analyses(4) on edible portion.~~
At least one sample from two onsite locations near the SITE BOUNDARY in different sectors.	Samples of ~~three different~~ kinds of broad ~~leaf vegeta-tion~~ grown ~~nearest each of two different~~ offsite ~~loca-tions of highest predicted annual average ground level D/Q if milk sampling is not performed (Ic10 - Ic13)~~.	Monthly during growing season.	Gamma isotopic(4) and I-131 analysis.
At least one sample from a control location at about 10 miles distant or beyond	~~One sample of each of the similar broad leaf vegeta-tion grown 15 to 30 km dis-tant in the least prevalent wind direction if milk sam-pling is not performed (Ic20 - Ic23).~~	~~Monthly during growing season.~~	~~Gamma isotopic(4) and I-131 analysis.~~

Each sample location will be designated by a number, name, or some other label.

Dry or inclement weather

TABLE 3.12-1 (Continued)

TABLE NOTATIONS

a point midway between the center of the two

(1) Specific parameters of distance and direction sector from ~~the centerline of one~~ reactors, and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978, and to Radiological Assessment Branch Technical Position, Revision 1, November 1979. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, ~~and~~ malfunction of ~~automatic~~ sampling equipment, or other justifiable reasons. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3. It is recognized that, at times, it may not be possible or practicable to continue to obtain adequate samples of the media of choice at the most desired location or time. In these instances, efforts shall be made to find suitable alternative media and locations ~~may be chosen~~ for the particular pathway in question and appropriate substitutions, if available, will be made within 30 days in the Radiological Environmental Monitoring Program. ~~given in the ODCM.~~ Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s), if any, with supporting information identifying the cause of the unavailability of samples for the pathway and justifying the selection of the new location(s) for obtaining samples, or the unavailability of suitable new locations

(2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation. ~~(The 40 stations is not an absolute number. The number of direct radiation monitoring stations may be reduced according to geographical limitations; e.g., at an ocean site, some sectors will be over water so that the number of dosimeters may be reduced accordingly. The frequency of analysis or readout for TLD systems will depend upon the characteristics of the specific system used and should be selected to obtain optimum dose information within minimal fading.)~~

(3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(5) The "upstream sample" shall be taken at a distance beyond significant influence of the discharge. The "downstream" sample shall be taken in an area beyond but near the mixing zone. ~~"Upstream" samples in an estuary must be taken far enough upstream to be beyond the plant influence. Salt water shall be sampled only when the receiving water is utilized for recreational activities.~~

(6) ~~A composite sample is one in which the quantity (aliquot) of liquid sampled is proportional to the quantity of flowing liquid and in which the method of sampling employed results in a specimen that is representative of the liquid flow. In this program~~ Composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.

~~(7) Groundwater samples shall be taken when this source is tapped for drinking or irrigation purposes in areas where the hydraulic gradient or recharge properties are suitable for contamination.~~

(7) ~~(8)~~ The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the ODCM.

~~(9) If harvest occurs more than once a year, sampling shall be performed during each discrete harvest. If harvest occurs continuously, sampling shall be monthly. Attention shall be paid to including samples of tuberous and root food products.~~

(8) A milking animal is a cow or goat producing milk for human consumption.

(9) If gamma isotopic analysis is not sensitive enough to meet the Lower Limit of Detection for I-131, a separate analysis for I-131 shall be performed.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	VEGETATION ~~FOOD PRODUCTS~~ (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-~~Nb~~-95	400				
Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			400	
~~Ba~~-La-140	200			~~300~~ 400	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[(1) (2)]

LOWER LIMIT OF DETECTION (LLD)[(3)]

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	~~FOOD~~ VEGETATION/PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000*					
Mn-54	~~15~~ 20		~~130~~ 100			
Fe-59	30		~~260~~ 300			
Co-58,~~60~~	~~15~~ 20		~~130~~ 100			
Co-60	20		100			
Zn-65	30		~~260~~ 300			
Zr-95	30					
~~Zr-Nb-95~~	~~15~~					
Nb-95	20					
I-131	1**	0.07		1	60	
Cs-134	~~15~~ 20	0.05	~~130~~ 100	~~15~~ 20	60	~~150~~ 200
Cs-137	~~18~~ 20	0.06	~~150~~ 200	~~18~~ 20	80	~~180~~ 200
Ba-140	60			60		
~~Ba~~-La-140	~~15~~ 20			~~15~~ 20		

*If no drinking water pathway exists, a value of 3000 pCi/l may be used.
~~**If no drinking water pathway exists, a value of 15 pCi/l may be used.~~

** LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS

(1) This list does not mean that only these nuclides as plant effluents are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3.

(2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.

(3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection (picoCuries per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide (sec^{-1}), and

Δt = the elapsed time between ~~environmental~~ sample collection, or end of the sample collection period, and time of counting (sec).

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3.

RADIOLOGICAL ENVIRONMENTAL MONITORING

Land within the Savannah River Plant may be excluded from this survey.

3/4.12.2 LAND USE ~~CENSUS~~ SURVEY

LIMITING CONDITION FOR OPERATION

if applicable

3.12.2 A Land Use ~~Census~~ Survey shall be conducted and shall identify within a distance of ~~8 km~~ (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. ~~[For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the Land Use Census shall also identify within a distance of 5 km (3 miles) the locations in each of the 16 meteorological sectors of all milk animals and all gardens of greater than 50 m² producing broad leaf vegetation.]~~

APPLICABILITY: At all times.

ACTION:

a. With a Land Use ~~Census~~ Survey identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.9.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use ~~Census~~ Survey identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) if samples are available. within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program. ~~after [October 31] of the year in which this Land Use Census was conducted.~~ Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

~~*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1, Part 4.c., shall be followed, including analysis of control samples.~~

by visual survey from automobile or aircraft

RADIOLOGICAL ENVIRONMENTAL MONITORING

SURVEILLANCE REQUIREMENTS

Survey

4.12.2 The Land Use ~~Census~~ shall be conducted during the growing season at least once per 12 months using that information that will provide ~~the best~~ good results, such as by a door-to-door survey, ~~aerial survey~~, ~~or~~ by consulting local agriculture authorities,. The results of the Land Use ~~Census~~ Survey shall be included in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3.

or by some combination of these methods as feasible.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission, that correspond to samples and analyses required by Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 The Interlaboratory Comparison Program shall be described in the ODCM. A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental ~~Operating~~ Surveillance Report pursuant to Specification 6.9.1.3.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.12

3.12.1, Footnote a:

This footnote was added to clarify that, whereas VEGP is a two-unit site, there will be one site-related radiological environmental monitoring program.

3.12.1, Action a:

This annual report, which documents radiological environmental monitoring activities for the calendar year, might more appropriately be entitled the Annual Radiological Environmental Surveillance Report; this title is used at our other nuclear plant. This change in terminology has been made throughout the remainder of the draft of Section 3/4.12.

3.12.1, Action b:

Confirmation provides assurance from making unnecessary reports. The word "confirmed" is added to be consistent with the Radiological Assessment Branch Technical Position (BTP), Rev. 1, November 1979.

The insertion of the words "from the end of the affected calendar quarter or after confirmation, whichever is later" clarifies when a report must be made.

3.12.1, Action c:

It needs to be clarified that this paragraph is not applicable to cases of temporary unavailability. It may not be possible to find suitable replacement locations for milk or vegetation (or other samples). Difficulty is being experienced in finding milk animals within 5 miles. Adequate vegetation samples may not be available during periods of hot, cold, or dry weather. Replacement locations can be added only when and if adequate samples are available. When samples become permanently or persistently unavailable at a given location, that sampling location is de facto deleted.

3.12.1, Footnote b:

This footnote was added to define the term "confirmed." The definition is consistent with the BTP referenced above.

Table 3.12-1:

General:

Designators for sample locations are provided in the ODCM and have been deleted from this table.

Distances are specified in miles as opposed to kilometers. The unit of miles is preferred since maps of the area around the plant are based on that unit. Wherever a range is specified for a sample location, the words "approximately" or "at about" have been inserted to achieve some needed flexibility.

Specific:

Footnote *:

This footnote was deleted on the basis that it simply provides guidance as to how this specification should be written. As such, it should not be a part of these Technical Specifications.

Item 1:

This footnote was deleted on the basis that it simply provides guidance as to how this specification should be written. As such, it should not be a part of these Technical Specifications.

Item 2, "Airborne":

The words "of the highest calculated annual average ground level D/Q" were deleted on the basis that there is not a large variation in the calculated values of the annual average D/Q at the site boundary among the azimuthal sectors. Practical considerations may prescribe the most suitable location as one other than that with the highest or next highest D/Q.

The words "as for example 15-30 Km distant and in the least prevalent wind direction" were deleted on the basis that they are instructional and should not be part of these Technical Specifications.

The insertion of the words "at about 10 miles distant or beyond" ensures that the D/Q value at the control station is significantly lower than at the indicator location.

Item 3, "Waterborne":

Groundwater sampling is not required on the basis that there are two distinct water zones underlying the plant site - an unconfined water table which is replenished by natural precipitation and a deep aquifer which is isolated from the water table by a thick aquiclude. The area on which the plant is situated is bounded by stream channels that act as drains for the

water table, thereby intercepting groundwater that moves laterally and prevents inflow or outflow to adjacent areas. (See VEGP Draft Environmental Statement Section 4.3.1.2.) Drinking water onsite is taken from the confined aquifer. Groundwater is not used for irrigation purposes within a few miles of the plant site.

At each downstream water treatment plant a composite sample of river water is taken near the intake and a grab sample of finished water is collected. We have been unable to obtain composite samples of finished water.

Item 4, "Ingestion":

a. Milk:

The insertion of the words "about 10 miles distant or beyond" will ensure that the D/Q value at the control station is significantly lower than that at the indicator location.

The words "when animals are on pasture, monthly at other times" were deleted since milk animals are on pasture essentially year round.

The addition of new note 9 eliminates the need for the I-131 analysis when the gamma isotopic analysis is sufficiently sensitive to meet the LLD for I-131. Otherwise, a separate analysis of I-131 will be performed.

The addition of new note 8 specifically defines a milk animal.

The deletion of a specific range for the control location for milking animals is based on the fact that milk animals are scarce and it is not practical to place unnecessary restrictions on control locations.

b. Fish:

The revisions to this portion of Table 3.12-1 are based on the following:

Only fish species are collected because sufficient quantities of invertebrates suitable for human consumption, such as oysters or clams, have not been found near the plant.

Not infrequently, it is difficult to obtain adequate samples of any commercially or recreationally important species of fish.

Sampling of seasonal fish is listed separately since sampling at the control station is not meaningful and the sampling periods may be indicated.

c. Food Products:

This was deleted on the basis that there is no downstream use of river water for irrigation purposes.

Grass may be substituted for leafy vegetation on the basis that grass has been found to be an effective sampling medium. Grass is available during a larger portion of the year than broad-leaf vegetation.

The basis for two onsite sample locations near the site boundary in different sectors is as follows:

> The BTP suggests the collection of vegetation at the site boundary is preferable to collections at an actual nearby offsite garden.
>
> There is not a large variation in the calculated values of the annual average D/Q at the site boundary amongst the azimuthal sectors. Practical considerations may prescribe the most suitable location as one other than that with the highest or next highest D/Q.
>
> The main advantages of maintaining an onsite grass or vegetation plot are:
>
> 1) One is not subject to the good will of a private citizen for obtaining adequate samples; and
>
> 2) The calculated annual average ground level D/Q is likely higher than any of the offsite gardens located in the annual land survey.

Concerning the sample from a control location at about 10 miles distant or beyond, the important thing is that the D/Q value at the control station should be significantly lower than at the indicator location. This may be assured by the 10-mile distance.

Table 3.12-1, Table Notation:

Note 1:

The addition of the words "a point midway between the center of the two" is based on the fact that all area maps have been constructed with the midway point as the origin. The centers of the two reactors are a few hundred feet apart.

The addition of the words "Each sample location will be designated by a number, name or some other label" was added to clarify that each location will be designated with a label in the ODCM.

The words "dry or inclement weather" were added to indicate that deviations may be necessitated as a result of the weather. The words "or other justifiable reasons" were added to allow for deviations resulting from unforeseen circumstances other than those explicitly called out.

Note 2:

The words deleted from this note were instructional in nature and as such should not be part of these Technical Specifications.

Note 5:

The deleted words are not applicable to VEGP.

Note 6:

It is impractical to obtain liquid samples which can be demonstrated to be proportional to flowrate in the Savannah River due to the complex nature of flow throughout a cross-section of the river under various flow conditions. True proportionality could only be achieved by relating the size of each alliquot to the integrated flowrate across the river cross-section at the time each aliquot is taken; this informaton cannot be obtained at reasonable cost and effort. However, because the Savannah River is protected from extreme flowrates by dams, representative sampling can reasonably be achieved by obtaining samples with automatic sampling equipment set to obtain aliquots at short intervals (on the order of hourly) with the sampling interval adjusted periodically in accordance with river flow conditions.

Note 7 (STS):

This note was deleted for the reasons already discussed under Item 3, "Waterborne."

Note 9 (STS):

This note was deleted on the basis that there is not downstream use of river water for irrigation purposes.

Table 3.12-2:

The separation of Zr-95 and Nb-95 and Cs-137 and Ba-140 removes any uncertainty regarding the reporting level or LLD.* The reporting levels for the specific radionuclides in the separated chains were derived on the same basis as the other radionuclides in Table 3.12-2. The reporting level is the concentration at which annual Appendix I dose limit to the most sensitive organ and age group would be acquired with 1 year's exposure.

Table 4.12-1:

The values in Table 4.12-1 are the same as those found in the BTP rounded to one significant figure. Laboratories usually report such results to one significant figure.

*Members of the decay chain may be determined separately by gamma isotopic analysis.

Note 1:

The addition of the words "as plant effluents" eliminates the need for reporting naturally occurring radionuclides which are not part of the plant effluents.

3/4.12.2:

The term "survey" is believed to be more appropriate than the word "census."

The statement pertaining to elevated releases was deleted on the basis that VEGP does not make elevated releases.

The addition of the statement excluding land within the Savannah River Plant from the survey is based on the fact that this land does not constitute a pathway to the public since access is controlled by SRP.

3/4.12.2, Action b:

The addition of the words "if samples are available" is based on the fact that, although milk and vegetation samples have generally been available upon request, it cannot be assumed that this will always be the case. For the VEGP site, no significance is recognized in delaying deletion until after October 31 since there is year round pasturing, and vegetation samples become scarce in hot, dry, and cold weather.

4.12.2:

The additions to this surveillance were made to provide clarification as to the manner in which the land use survey may be conducted.

3.12.2, Footnote*:

This footnote was deleted on the basis that, although grass or leafy vegetation is to be sampled at two site boundary locations, the garden survey is to be performed for dose calculations due to plant effluents.

3/4.12.3:

The words "and analyses" were inserted to clarify that only those analyses of applicable samples as required by Table 3.12-1 need be performed in the Interlaboratory Comparison Program.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4. In the event of a disagreement between the requirements stated in these Technical Specifications and those stated in an applicable Federal Regulation or Act, the requirements stated in the applicable Federal Regulation or Act shall take precedence and shall be met.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 The specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both the required ECCS subsystems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one Spray System is inoperable. Under the requirements of Specification 3.0.3, if both the required Containment Spray Systems are inoperable, within 1 hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours. It is acceptable to initiate and complete a reduction in OPERATIONAL MODES in a shorter time interval than required in the ACTION statement and to add the unused portion of this allowable out-of-service time to that provided for operation in subsequent lower OPERATION MODE(S). Stated allowable out-of-service times are applicable regardless of the OPERATIONAL MODE(S) in which the inoperability is discovered but the times provided for achieving a mode reduction are not applicable if the inoperability is discovered in a mode lower than the applicable mode. For example if the Containment Spray System was discovered to be inoperable while in STARTUP, the ACTION Statement would allow up to 156 hours to achieve COLD SHUTDOWN. If HOT STANDBY is attained in 16 hours rather than the allowed 78 hours, 140 hours would still be available before the plant would be required to be in COLD SHUTDOWN However, if this system was discovered to be inoperable while in HOT STANDBY, the 6 hours provided to achieve HOT STANDBY would not be additive to the time available to achieve COLD SHUTDOWN so that the total allowable time is reduced from 156 hours to 150 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with: (1) the full complement of required systems, equipment, or components OPERABLE and (2) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out-of-service provisions contained in the ACTION statements.

The intent of this provision is to ensure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

4.0.1 This specification provides that surveillance activities necessary to ensure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over three test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under these criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE when such items are found or known to be inoperable although still meeting the Surveillance Requirements. ~~Items may be determined inoperable during use, during surveillance tests, or in accordance with this specification. Therefore, ACTION statements are entered when the Surveillance Requirements should have been performed rather than at the time it is discovered that the tests were not performed.~~

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant STARTUP or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to 1 week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

In MODES 1 and 2

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~[1.6%]~~ 1.3% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. ~~With T_{avg} less than 200°F, the reactivity transients resulting from a postulated steam line break cooldown are minimal and a 1% $\Delta k/k$ SHUTDOWN MARGIN provides adequate protection.~~

Replace with insert to B 3/4 1-1.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

Insert A to B 3/4 1-1

In MODES 3, 4, and 5 the most restrictive condition occurs at BOL, associated with a Boron Dilution Accident. In the analysis of this accident a minimum SHUTDOWN MARGIN of (Later) delta-k/k (MODES 3 and 4) and (Later) delta-k/k (MODE 5) is required to allow the operator 15 minutes from the initiation of the Source Range High Flux at Shutdown Alarm to reactor criticality. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting requirement and is consistent with the FSAR accident analysis assumptions.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value [-3.9] x 10^{-4} $\Delta k/k/^{\circ}F$. The MTC value of [-3.0] x 10^{-4} $\Delta k/k/^{\circ}F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of [-3.9] x 10^{-4} $\Delta k/k/^{\circ}F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

551

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than ~~[541]~~°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, ~~(3) the P-12 interlock is above its setpoint,~~ 3 (~~4~~) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (~~5~~) 4 the reactor vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, and (4) the boric acid transfer pumps~~, (5) associated Heat Tracing Systems, and (6) an emergency power supply from OPERABLE diesel generators~~.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN

of 1.3% delta-k/k for MODES 1 and 2 and of (Later) delta-k/k for MODES 3 and 4.

(Later)

BORATION SYSTEMS (Continued)

MARGIN ~~from expected operating conditions of 1.6% Δk/k~~ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires ~~[5106]~~ gallons of [7000] ppm borated water from the boric acid storage tanks or ~~[52,622]~~ gallons of 2000 ppm borated water from the refueling water storage tank (RWST). usable volume (Later)

With the RCS temperature below 200°F, one Boron Injection System is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single Boron Injection System becomes inoperable.

~~The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below [275]°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.~~

(Later) (Later)

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of ~~1%~~ Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either __ gallons of [7000] ppm borated water from the boric acid storage tanks or (Later) gallons of 2000 ppm borated water from the RWST. usable volume

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between [8.5] and ~~[11.0]~~ 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

~~3/4.1.3 MOVABLE CONTROL ASSEMBLIES~~

~~The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits. Verification that the Digital Rod Position Indicator agrees with the demanded position within ± 12 steps at 24, 48, 120,~~

MOVABLE CONTROL ASSEMBLIES (Continued)

and 228 steps withdrawn for the Control Banks and 18, 210, and 228 steps withdrawn for the Shutdown Banks provides assurances that the Digital Rod Position Indicator is operating correctly over the full range of indication. Since the Digital Rod Position Indication System does not indicate the actual shutdown rod position between 18 steps and 210 steps, only points in the indicated ranges are picked for verification of agreement with demanded position.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to [541]°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a Reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section are necessary to ensure that the following requirements are met at all times during normal operation. By observing that the RCCAs are positioned above their respective insertion limits during normal operation,

1. At any time in life for MODE 1 and 2 operation, the minimum SHUTDOWN MARGIN will be maintained. For operational MODES 3, 4, 5, and 6, the reactivity condition consistent with other specifications will be maintained with all RCCAs fully inserted by observing that the boron concentration is always greater than an appropriate minimum value.

2. During normal operation the enthalpy rise hot channel factor, $F_{\Delta H}$, will be maintained within acceptable limits.

3. The consequences of an ejected RCCA accident will be restricted below the limiting consequences referred to in the ejected rod analysis.

4. The core can be made subcritical by the required SHUTDOWN MARGIN with one RCCA stuck. In the event of an RCCA ejection, the core can be made subcritical with two RCCAs stuck, where one of the RCCAs is assumed to be the worst ejected rod control assembly.

5. The trip reactivity assumed in the accident analysis will be available.

6. Dropping an RCCA into the core or statically misaligning an RCCA during normal operation will not violate the thermal design basis with respect to DNBR.

7. The uncontrolled withdrawal of an RCCA will result in consequences no more severe than presented in the accident analysis.

8. The uncontrolled withdrawal of a control assembly bank will not result in a peak power density that exceeds the center line melting criterion.

OPERABILITY of the control rod position indicator channels (LCO 3.1.3.2) is required to determine control rod positions and thereby ensure compliance with the control rod alignment.

OPERABILITY of the Demand Position Indication System (LCO 3.1.3.2) is required to determine bank demand positions and thereby ensure compliance with the insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that some of the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER, either of these restrictions provide assurance of fuel rod integrity during continued operation provided no further abnormal condition develops.

For Specification 3.1.3.1 ACTIONS b and c it is incumbent upon the plant to verify the trippability of the inoperable control rod(s). This may be by verification of a control system failure, usually electrical in nature, or that the failure is associated with the control rod stepping mechanism. In

the event the plant is unable to verify the rod(s) trippability, it must be assumed to be untrippable and thus fall under the requirements of ACTION a. Assuming a controlled shutdown from 100% RATED THERMAL POWER, this allows approximately four hours for this verification.

The maximum rod drop time permitted by (LCO 3.1.3.4) is consistent with the assumed rod drop time used in the accident analyses. Measurement with T_{avg} 551 degrees-F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Bank demand positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is INOPERABLE. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of ~~2.32~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes. 2.30

Target flux difference is determined at equilibrium xenon conditions. The ~~full-length~~ rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1-hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure ~~(3.2-1)~~ while at THERMAL POWER levels between 50% and 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the 1-minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two or more OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% and 90% and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band.

~~3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR~~

~~The limits on heat flux hot channel factor, RCS flow rate, and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.~~

~~Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:~~

~~a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;~~

~~b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;~~

FIGURE B 3/4 2-1
TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

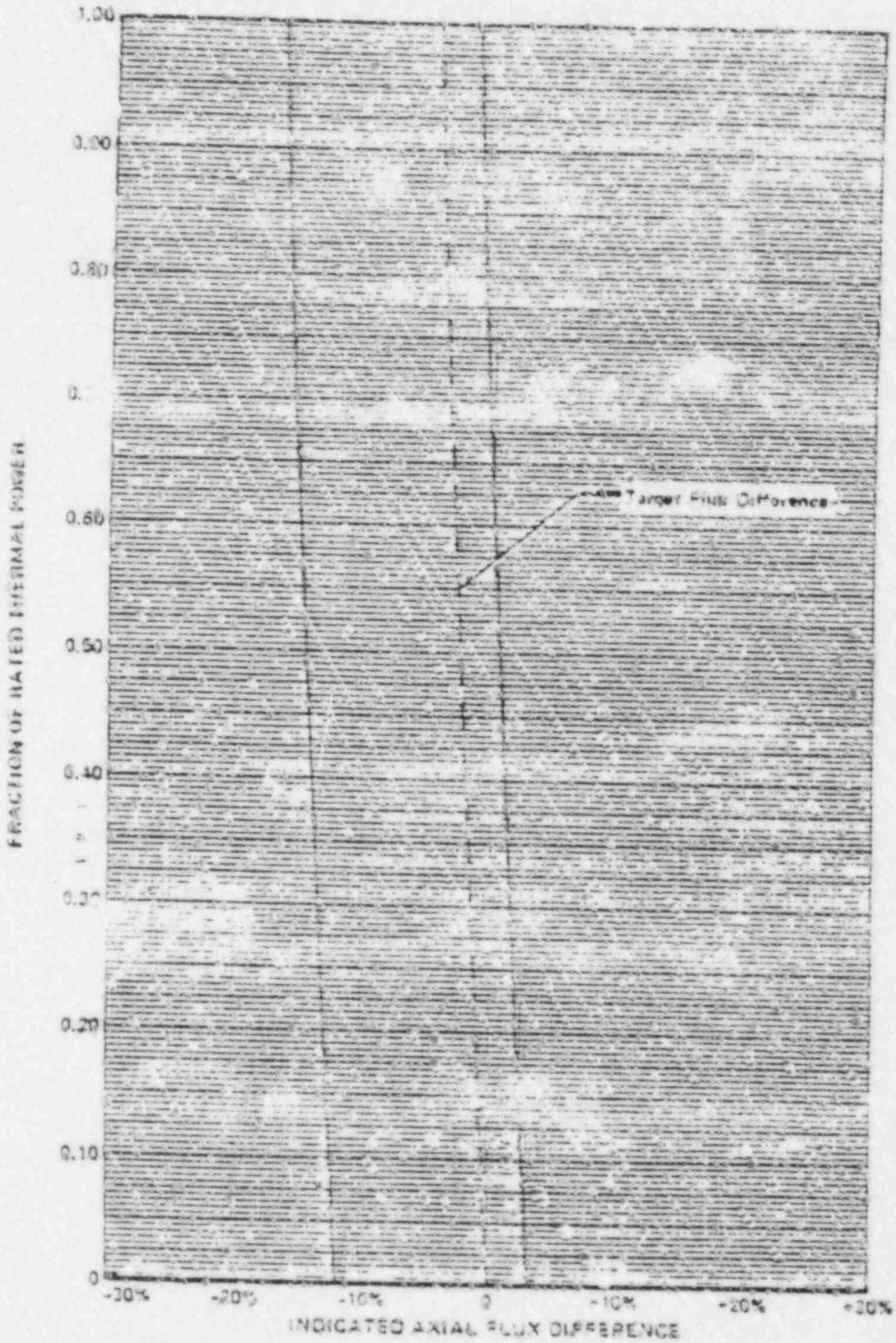


FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following:

a. Design limit DNBR of [1.30 vs 1.28],

b. Grid Spacing (K_s) of [0.046 vs 0.059],

c. Thermal Diffusion Coefficient of [0.038 vs 0.059],

d. DNBR Multiplier of [0.86 vs 0.88], and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, NUCLEAR ENTHALPY HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measureable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position.

b. Control rod banks are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specification 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial shape for all permissible rod insertion limits.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full core map taken with the incore detector flux mapping system and a 3% allowance is appropriate for manufacturing tolerance.

When $F^N_{\Delta H}$ is measured, (i.e., inferred), measurement uncertainty (i.e., the appropriate uncertainty on the incore inferred hot rod peaking factor) must be allowed for and 4% is the appropriate allowance for a full core map taken with the incore detection system.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28,
2) Grid Spacing (K_s) of 0.046 vs. 0.059,
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059,
4) DNBR Multiplier of 0.86 vs. 0.88, and
5) Pitch reduction

The applicable values of rod bow penalties are referenced in the FSAR.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of [2.1]% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

The measurement error for RCS total flow rate is based upon performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi which might not be detected could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi is included in Figure 3.2-3. Any fouling which might bias the RCS flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken before performing subsequent precision heat balance measurements, i.e., either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation shown on Figure 3.2-3.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The radial peaking factor, $F_{xy}(Z)$, is measured periodically to provide additional assurance that the hot channel factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor limit report per Specification 6.9.1.6 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plant power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are H-3, H-13, C-8, N-8, E-5, E-11, L-5, and L-11.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of (Later) and the indicated pressurizer pressure value of (Later) psig correspond to analytical limits of (Later) and (Later) psig, respectively, with allowance for measurement uncertainty.

QUADRANT POWER TILT RATIO (Continued)

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient. The indicated T_{avg} value of [581]°F and the indicated pressurizer pressure value of [2220] psig correspond to analytical limits of 595°F and 2205 psig respectively, with allowance for measurement uncertainty.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

BASES

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of the flow rate on a 12 hour basis.

consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation,

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic ~~is maintained~~, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. See Insert for B 3/4 3-1.

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 3.3-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 3.3-4, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span. R or Rack Error is the "as measured" deviation, in the percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of

Insert to Page B 3/4 3-1

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System", and supplements to that report. Surveillance intervals and out of service times were determined based upon maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

the sensor from its calibration point or the value specified in Table 3.3-4, in percent span, from the analysis assumptions. Use of Equation 3.3-1 allows for a sensor draft factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the Reactor trip and the Engineered Safety Features actuation associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either: (1) in place, onsite, or offsite test measurements, or (2) utilizing replacement sensors with certified response time.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) ~~Safety Injection~~ ECCS pumps start and automatic valves position, (2) Reactor trip, (3) feed water isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) auxiliary feedwater pumps start and automatic valves position, (10) containment ~~cooling fans~~ fan coolers start and automatic valves position, (11) ~~essential service~~ Nuclear Service cooling and component cooling water pumps start and automatic valves position, and (12) ~~Control Room Isolation and Ventilation~~ Control Room Ventilation Emergency Actuation Systems start.

REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically and low steam line pressure reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure, and low steam line pressure and allows steam line isolation on negative steam line pressure rate to become active upon manual block of low steam line pressure SI.

~~P-12 On increasing reactor coolant loop temperature, P-12 automatically reinstates Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure, and provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 allows the manual block of Safety Injection actuation on high steam flow coincident with either low-low T_{avg} or low steam line pressure and automatically removes the arming signal from the Steam Dump System.~~

P-14 On increasing steam generator water level, P-14 automatically trips all feedwater isolation valves, initiates a turbine trip, and inhibits feedwater control valve modulation.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance. The radiation monitors for plant operations senses radiation levels in selected plant systems and locations and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM

The OPERABILITY of the Remote Shutdown System ensures that sufficient capability is available to permit safe shutdown of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the Remote Shutdown System ensures that a fire will not preclude achieving safe shutdown. The Remote Shutdown System instrumentation,

REMOTE SHUTDOWN SYSTEM (Continued)

control, and ~~power circuits and~~ transfer switches necessary to eliminate effects of the fire and allow operation of instrumentation, and control ~~and power~~ circuits required to achieve and maintain a safe shutdown condition are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and ~~Appendix R to 10 CFR Part 50.~~ CMEB 9.5.1.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, ~~Revision 3,~~ "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," ~~May 1983~~ Revision 2, December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the Chlorine Detection Systems ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, ~~Revision 1,~~ "Protection of Nuclear Power Plant Control room Operators Against an Accidental Chlorine Release," ~~January 1977~~ February 1975.

~~3/4.3.3.8 FIRE DETECTION INSTRUMENTATION~~

~~The OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.~~

~~Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant's Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.~~

~~The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for~~

FIRE DETECTION INSTRUMENTATION (Continued)

~~any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY~~.

3/4.3.3.~~9~~ 8 LOOSE PART DETECTION SYSTEM

The OPERABILITY of the Loose-Part Detection System ensures that sufficient capability is available to detect loose metallic parts in the Reactor System and avoid or mitigate damage to Reactor System components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.~~10~~ 9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.~~11~~ 10 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1 x 10^{-6} μCi/ml are measurable.

3/4.3.3.11 - See insert for B 3/4 3-6.

INSERT FOR B 3/4 3-6

3/4.3.3.12 HIGH ENERGY LINE BREAK ISOLATION SENSORS

The operability of the high energy line break isolation sensors ensures that the capability is available to promptly detect and initiate protective action in the event of a line break. This capability is required to prevent damage to safety-related systems and structures in the auxiliary building.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety-related components, equipment or structures.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip System breakers. ~~Single failure considerations require that two loops be OPERABLE at all times~~.

In MODE 4, and in MODE 5 train with reactor coolant loops filled, a single reactor coolant loop or RHR ~~loop~~ provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two ~~loops~~ loops/trains (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR ~~loop~~ train provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR ~~loops~~ trains be OPERABLE. Insert to B 3/4 4-1 here.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to ~~[275]~~ 350°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the nominal 10 effective full power years (EFPY) Appendix G reactor vessel NDT limits ~~limits of Appendix G by either: (1) restricting the water volume in the pressurizer and thereby providing a volume for the reactor coolant to expand into, or (2)~~ by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

~~[OPTIONAL]~~

~~The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the stop valves provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its stop valves ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications~~.

Insert 1 to B 3/4 4-1

The locking closed of the required valves in Mode 5 (with the loops not filled) precludes the possibility of uncontrolled boron dilution of the filled portion of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the boron dilution accident in the safety analysis.

~~REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)~~

~~[OPTIONAL]~~

~~Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water-induced reactivity transient~~.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. ~~at the valve Setpoint.~~ The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR ~~loop~~, train connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Cold Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may ~~likely~~ result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System primary (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair or Plugging will be required for all tubes with imperfections exceeding the plugging limit of [40]% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

STEAM GENERATORS (Continued)

10 CFR 50.73

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission ~~in a Special Report~~ pursuant to ~~Specification 6.9.2 within 30 days~~ and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators ~~not isolated from the RCS~~ ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.

~~The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating~~

OPERATIONAL LEAKAGE (Continued)

~~valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses~~.

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. ~~It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required~~. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an

SPECIFIC ACTIVITY (Continued)

appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state primary ~~reactor~~-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Vogtle site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than ~~10~~ 14 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of ~~10~~ 14 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes~~, which allows a distinction between the radionuclides above and below a half-life of 10 minutes~~. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

PRESSURE/TEMPERATURE LIMITS (Continued)

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than [625]°F, and auxiliary

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Prior to each test, The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the ~~1976 Summer~~ 1972 Addenda to Section III of the ASME Boiler and Pressure Vessel Code. ~~and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.~~

See insert for B 3/4 4-8

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of [12] effective full power years (EFPY) of service life. The ~~[12]~~ 16 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphorus content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by either: Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of ~~[12]~~ 16 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Insert for B 3/4 4-8

The heatup and cooldown limit curves in Figures 3.4-2 and 3.4-3 are applicable to Vogtle-Unit 1 for up to 16 EFPY. The most limiting material for Vogtle-Unit 1 has an initial RT_{NDT} of 30°F and a copper content of 0.06 WT %, whereas the heatup and cooldown curves in Figures 3.4-2 and 3.4-3 are based on an initial RT_{NDT} of 40°F and a copper content of 0.10 WT %. As a result, applicable to Vogtle-Unit 1.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	ASME MATERIAL TYPE	CU %	P %	NDTT °F	50 FT-LB/35 MIL TEMP °F LONG	50 FT-LB/35 MIL TEMP °F TRANS	RT_{NDT} °F	MIN. UPPER SHELF FT-LB LONG	MIN. UPPER SHELF FT-LB TRANS

TABLE B 3/4.4-1

VOGTLE UNIT NO. 1 REACTOR VESSEL MATERIAL PROPERTIES

Component	Code No.	Mat'l Spec No.	Cu (%)	P (%)	NDT (°F)	50-ft-lb 35 mil Temp (°F)	RT_{NDT} (°F)	USE NMWD* (ft-lb)	USE MWD** (ft-lb)
Closure Head Dome	B8807-1	A533BCL.1	.16	.008	-50	75	15	88	-
Closure Head Torus	B8808-1	A533BCL.1	.14	.010	-30	68	8	85	-
Closure Head Flange	B8801-1	A508CL.2	-	.011	20	<40	20	132	-
Vessel Flange	B8802-1	A508CL.2	-	.014	0	<60	0	119	-
Inlet Nozzle	B8809-1	A508CL.2	-	.011	-20	<10	-20	107	-
Inlet Nozzle	B8809-2	A508CL.2	-	.014	-10	<50	-10	95	-
Inlet Nozzle	B8809-3	A508CL.2	-	.013	-10	<10	-10	117	-
Inlet Nozzle	B8809-4	A508CL.2	-	.014	-20	<10	-20	105	-
Outlet Nozzle	B8810-1	A508CL.2	-	.006	-10	<50	-10	>124	-
Outlet Nozzle	B8810-2	A508CL.2	-	.006	-10	<50	-10	>100	-
Outlet Nozzle	B8810-3	A508CL.2	-	.006	-10	<50	-10	>102	-
Outlet Nozzle	B8810-4	A508CL.2	-	.006	-10	<10	-10	> 75	-
Nozzle Shell	B8804-1	A533BCL.1	.14	.011	-10	88	28	94	-
Nozzle Shell	B8804-2	A533BCL.1	.10	.006	-40	75	15	104	-
Nozzle Shell	B8804-3	A533BCL.1	.14	.013	-30	100	40	92	-
Inter. Shell	B8805-1	A533BCL.1	.08	.004	0	60	0	90	-
Inter. Shell	B8805-2	A533BCL.1	.08	.004	-10	80	20	100	-
Inter. Shell	B8805-3	A533BCL.1	.06	.003	-20	90	30	107	-
Lower Shell	B8606-1	A533BCL.1	.05	.005	-50	80	20	116	-
Lower Shell	B8606-2	A533BCL.1	.05	.009	-10	80	20	113	-
Lower Shell	B8606-3	A533BCL.1	.06	.007	-20	70	10	118	-
Bottom Head Torus	B8813-1	A533BCL.1	.13	.009	-40	50	-10	88	-
Bottom Head Dome	B8812-1	A533BCL.1	.10	.009	-40	32	-28	122	-
Inter & Lower Shell Vent Weld Seams and Girth Seam	GT.43	SAW	.03	.007	-80	<20	-80	>129	-

* Normal to major working directions.
** Major working direction.

FIGURE B 3/4.4-1

FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE

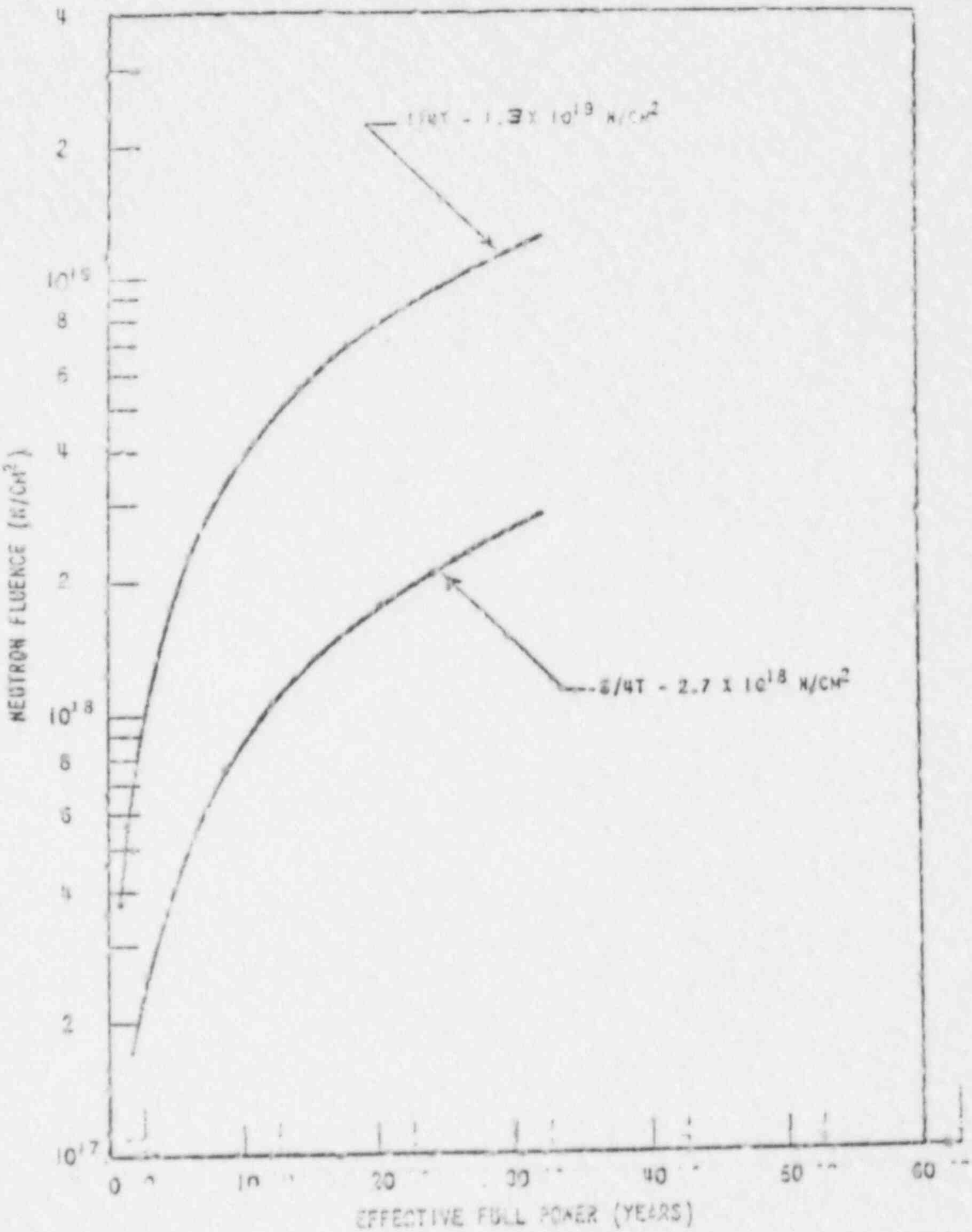


Figure B 3/4.4-1. Fast Neutron Fluence (E > 1 MeV) as a Function of Full Power Service Life

FIGURE B 3/4.4-2

EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSELS EXPOSED TO 550°F

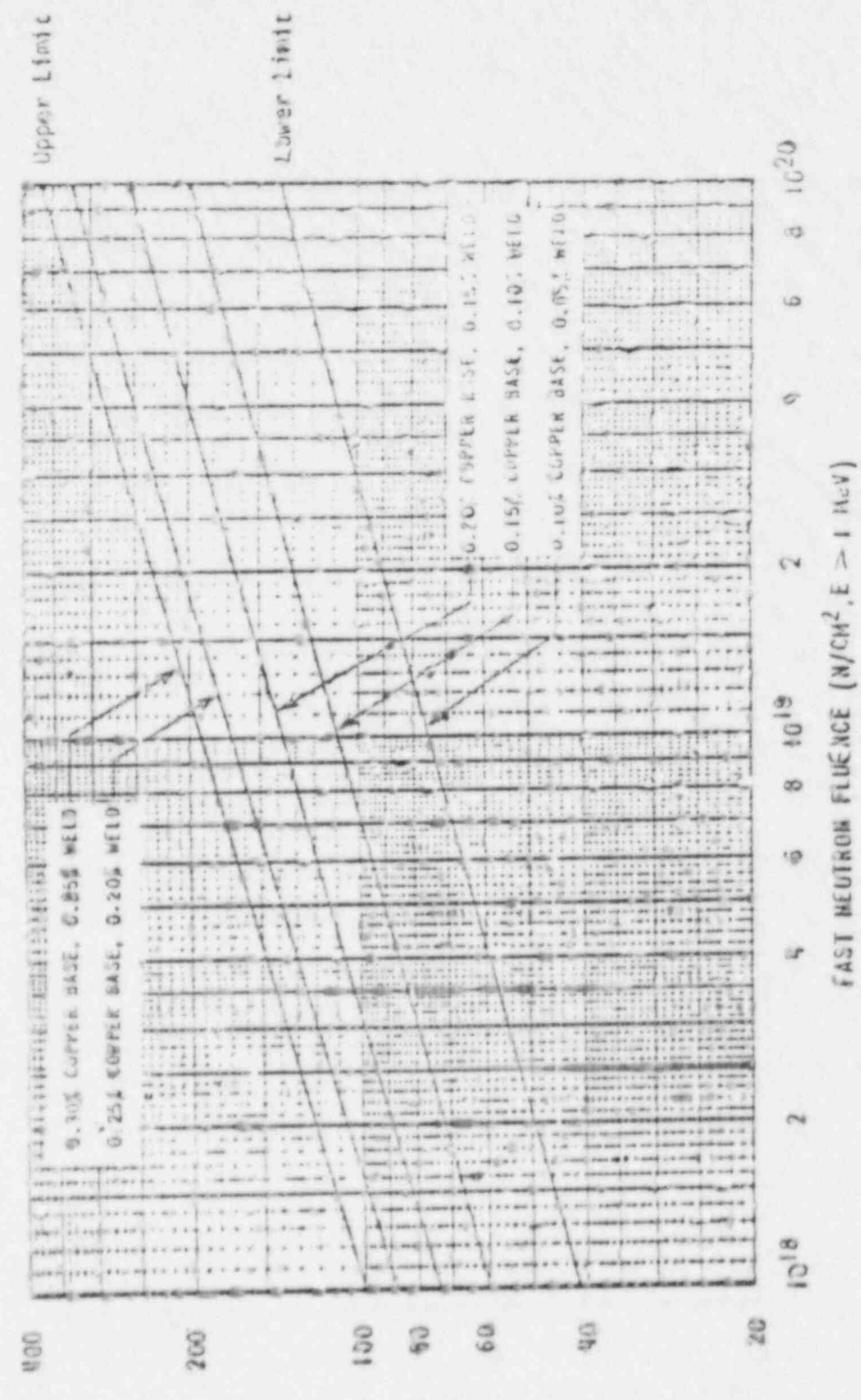


Figure B 3/4.4-2. Effect of Fluence and Copper on Shift and RT_{NDT} for Reactor Vessel Steels Exposed to Irradiation at 550°F

PRESSURE/TEMPERATURE LIMITS (Continued)

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of 82 ASTM E185-~~73~~ and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in ~~WCAP-7924-A~~. *the following paragraphs.*

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

PRESSURE/TEMPERATURE LIMITS (Continued)

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location

PRESSURE/TEMPERATURE LIMITS (Continued)

for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued)

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

next
~~Finally~~, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

See insert for B 3/4 4-15 → Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

COLD
~~LOW TEMPERATURE~~ OVERPRESSURE PROTECTION SYSTEMS

the RHR suction relief valves
350

The OPERABILITY of two PORVs, or an RCS vent ~~opening of at least square inches~~ ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to ~~[275]~~°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50 °F above the RCS cold leg temperatures, or (2) the start of ~~a HPSI pump and its~~ all three charging pumps and subsequent injection into a water-solid RCS.

COPS Cold
The Maximum Allowed PORV Setpoint for the ~~Low Temperature~~ Overpressure Protection Systems ~~(LTOPS)~~ COPS is derived by analysis which models the performance of the ~~LTOPS~~ assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that ~~Appendix G~~ criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all ~~but one~~ safety injection pumps ~~and all but one centrifugal charging pump~~ while in MODES 4, 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

COPS
The Maximum Allowed PORV Setpoint for the ~~LTOPS~~ will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

the nominal 10 effective full power years (EFPY) Appendix G

Insert for B 3/4 4-15

Finally, the new 10CFR50 Appendix G Rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. This rule states that the minimum metal temperature of the closure flange regions should be at least 120^{o}F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig for Westinghouse Plants). For Vogtle Unit 1 the minimum temperature of the closure flange and vessel flange regions is 140^{o}F, since the limiting RT_{NDT} is 20^{o}F (see Table B 3/4-4.1). The Vogtle Unit 1 heatup curve shown on Figure 3-4.2 is not impacted by the new 10CFR50 rule. However, the Vogtle Unit 1 cooldown curve shown in Figure 3-4.3 is impacted by the new 10CFR50 rule.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, ~~____ Edition and Addenda~~ through the 1970 Winter Addenda.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System Vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of ~~least one Reactor Coolant System~~ a reactor vessel head vent path ~~from the [reactor vessel head], the [Reactor Coolant System high point], the [pressurizer steam space], and the [isolation condenser high point]~~ ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for all ~~a maximum of one centrifugal charging pump and one~~ safety injection pumps to be Inoperable ~~OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable below [275]°F~~ below 350°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, ~~and~~ (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses, and (4) to ensure that centrifugal charging pump injection flow which is directed through the seal injection path is less than or equal to the amount assumed in the safety analysis.

~~3/4.5.4 BORON INJECTION SYSTEM [OPTIONAL]~~

The OPERABILITY of the ~~Boron Injection System~~ Refueling Water Storage Tank (RWST) as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

~~The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the steam line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.~~

~~[The OPERABILITY of the redundant heat tracing channels associated with the boron injection system ensures that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 22,500 ppm boron.]~~

3/4.5.~~5~~4 REFUELING WATER STORAGE TANK

~~The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS~~ In the event of a LOCA, The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the

REFUELING WATER STORAGE TANK (Continued)

RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between [8.5] and ~~[11.0]~~ 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a ~~or 0.75 L_t, as applicable,~~ during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

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

~~3/4.6.1.4 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS [OPTIONAL]~~

~~The OPERABILITY of the Isolation Valve and Containment Channel Weld Pressurization Systems is required to meet the restrictions on overall containment leak rate assumed in the safety analyses. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix J of 10 CFR Part 50~~.

3/4.6.1.~~5~~ 4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of ~~[3]~~ psig, and (2) the containment peak pressure does not exceed the design pressure of ~~[54]~~ 52 psig during [~~LOCA or~~ steam line break conditions].

The maximum peak 41.9 pressure assuming an initial containment pressure of 0.3 psig expected to be obtained from a [~~LOCA or~~ steam line break] event is ~~[45]~~ psig. The limit of ~~[3]~~ psig for initial positive containment pressure will limit the total pressure to ~~[40]~~ 45 psig, which is less than design pressure and is consistent with the safety analyses.

3/4.6.1.~~6~~ 5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the safety analysis for a [~~LOCA or~~ steam line break accident]. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.~~7~~ 6 CONTAINMENT STRUCTURAL INTEGRITY

~~[Prestressed concrete containment with ungrouted tendons]~~

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~[40]~~ 41.9 psig in the event of a ~~[LOCA or~~ steam line break accident~~]~~. The measurement of containment tendon lift-off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. (The tendon ~~wire or~~ strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)

The Surveillance Requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of ~~proposed~~ Revision 2 of Regulatory Guide 1.35, "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures," ~~April 1979~~, and proposed Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," April 1979.

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results of the engineering evaluation, and the corrective actions taken.

Sealed closed isolation valves are isolation valves under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, the use of blind flanges, or removal of power to the valve operator.

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

~~[Reinforced concrete containment]~~

~~This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of [48] psig in the event of a [LOCA or steam line break accident]. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability~~.

3/4.6.1.~~8~~7 CONTAINMENT VENTILATION SYSTEM

The ~~[42~~24-inch~~]~~ containment purge supply and exhaust isolation valves are required to be sealed closed during plant operations since these valves have not been demonstrated capable of closing during a ~~[~~LOCA or steam line break accident~~]~~. Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are sealed closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the ~~[8~~14-inch~~]~~ purge supply and exhaust isolation valves ~~since, unlike the [42-inch] valves, the [8-inch] valves are capable of closing during a [LOCA or steam line break accident]. Therefore,~~ to ensure that the SITE BOUNDARY dose guideline of 10 CFR Part 100 would not be exceeded in the event of an accident during containment PURGING operation. ~~Operation with one pair of these valves open will be limited to [1000] hours during a calendar year~~. ~~The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience. Only safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities, may be used to support the additional time requests. Only safety-related reasons should be used to justify the opening of these isolation valves during MODES 1, 2, 3, and 4 in any calendar year regardless of the allowable hours~~.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a ~~[~~LOCA or steam line break~~]~~. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

CONTAINMENT SPRAY SYSTEM (Continued)

~~[Credit taken for iodine removal]~~

The Containment Spray System and the Containment Cooling System, both provide ~~are redundant to each other in providing~~ post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

~~[No credit taken for iodine removal]~~

~~The Containment Spray System and the Containment Cooling System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Spray System and Containment Cooling System have been interrelated and adjusted to reflect this additional redundancy in cooling capability.~~

3/4.6.2.2 SPRAY ADDITIVE SYSTEM ~~[OPTIONAL]~~

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between ~~[~~8.5~~]~~ and ~~[11.0]~~ 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM ~~[OPTIONAL]~~

The OPERABILITY of the Containment Cooling System ensures that: (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

~~[Credit taken for iodine removal by spray systems]~~

The Containment Cooling System and the Containment Spray System, both provide ~~are redundant to each other in providing~~ overlapping post-accident cooling of the containment atmosphere. As a result of this ~~redundancy in~~ cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. However, the allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere.

~~[No credit taken for iodine removal by spray systems]~~

~~The Containment Cooling System and the Containment Spray System are redundant to each other in providing post-accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the Containment Spray System, the allowable out-of-service time requirements for the Containment Cooling System and Containment Spray System have been interrelated and adjusted to reflect this additional redundancy in cooling capacity.~~

~~3/4.6.3 IODINE CLEANUP SYSTEM [OPTIONAL]~~

~~The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. Operation of the system with the heaters operating for at least 10 hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. ANSI N510-1975 and Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems" are used as procedural guides for surveillance testing.~~

3/4.6.~~4~~ 3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistant with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.~~5~~ 4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit [or the Purge System] is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. ~~[Cumulative operation of the Purge System with the heaters operating for 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters].~~ These Hydrogen Control Systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

~~3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL]~~

~~The OPERABILITY of the Penetration Room Exhaust Air Cleanup System ensures that radioactive materials leaking from the containment atmosphere through containment penetrations following a LOCA are filtered and adsorbed prior to reaching the environment. Operation of the system with the heaters operating for at~~

PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL] (Continued)

least 10 continous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the LOCA analyses. ANSI N510-1975 and Generic Letter 83-13, "Clarification of Surveillance Requirements for HEPA Filters and Charcoal Adsorber Units in Standard Technical Specifications on ESF Cleanup Systems" are used as procedural guides for surveillance testing.

3/4.6.7 VACUUM RELIEF VALVES [OPTIONAL]

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than ___ psig. This condition is necessary to prevent exceeding the containment design limit for internal vacuum of ___ psig.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the 1185 main steam line Code safety valves 1304 ensures that the Secondary System pressure will be limited to within 110% ~~[1100~~ psig~~]~~ of its design pressure of ~~[1000]~~ psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified 18,607,220 valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is ____ lbs/h which is 123 % of the total secondary steam flow of ____ lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

15,140,000

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following ~~bases:~~ basis:

For ~~N~~ 4 loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

~~For N-1 loop operation~~

$$\cancel{SP = \frac{(X) - (Y)(U)}{X} \times (76)}$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

~~U = Maximum number of inoperable safety valves per operating steam line,~~

SAFETY VALVES (Continued)

[109] = Power Range Neutron Flux-High Trip Setpoint for ~~[N]~~ 4 loop operation,

~~[76] = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for [N-1] loop operation,~~

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than [350]°F from normal operating conditions in the event of a total loss-of-offsite power.

Replace with insert to B 3/4 7-2

~~Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of [350] gpm at a pressure of [1133] psig to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of [700] gpm at a pressure of [1133] psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than [350]°F when the Residual Heat Removal System may be placed into operation~~.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

, followed by a cooldown to RHR initiation conditions

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor-to-secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

Insert to Page B 3/4 7-2

The auxiliary feedwater system is capable of delivering a total feedwater flow of 510 gpm at a pressure of 1221 psig to the entrance of at least two steam generators while allowing for: 1) any possible spillage through the design worst case break of the main feedwater line: 2) the design worst case single failure and, 3) recirculation flow (applicable for turbine-driven auxiliary feedwater pump only). This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F, at which point the Residual Heat Removal System may be placed in operation. Because it is not desirable to inject cold auxiliary feedwater into the steam generators during power generation, the pumps must be tested on miniflow (recirculation). The surveillance acceptance criteria are based on the miniflow testing configuration and are specified to ensure the above limits are met during injection to the steam generators.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of ~~[~~70~~]~~°F and ~~[~~200~~]~~ psig are based on a steam generator RT_{NDT} of 60 °F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 NUCLEAR SERVICE COOLING ~~SERVICE~~ WATER SYSTEM

The OPERABILITY of the Nuclear Service Cooling ~~Service~~ Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK ~~[OPTIONAL]~~

The limitations on the ultimate heat sink level, ~~and~~ temperature and minimum required number of OPERABLE fans ensure that sufficient cooling capacity is available to either: (1) provide normal cooldown of the facility or (2) mitigate the effects of accident conditions within acceptable limits.

ULTIMATE HEAT SINK (Continued)

and minimum required number of OPERABLE fans

The limitations on minimum water level, ~~and~~ maximum temperature are based on providing an adequate ~~a 30-day~~ cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

~~3/4.7.6 FLOOD PROTECTION [OPTIONAL]~~

~~The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation ____ Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety-related equipment.~~

3/4.7.6 ~~7~~ CONTROL ROOM EMERGENCY FILTRATION ~~AIR CLEANUP~~ SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system during accident conditions, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. Operation of the system with the heaters control circuit energized ~~operating~~ for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing.

3/4.7.7 ~~8~~ PIPING PENETRATION AREA FILTRATION AND EXHAUST SYSTEM ~~ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM~~

The OPERABILITY of the Piping Penetration Area Filtration and Exhaust System ~~ECCS Pump Room Exhaust Air Cleanup System~~ ensures that radioactive materials leaking from the containment mechanical penetration rooms and ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. Operation of the system with the heaters control circuit energized ~~operating~~ for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the safety analyses. ANSI N510-~~1975~~ 1980 will be used as a procedural guide for surveillance testing.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the [Unit Review Group]. The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures on a given system and is determined by the number of inoperable snubbers found during an inspection of each system. In order to establish the inspection frequency for each type of snubber on a safety-related system, it was assumed that the frequency of snubber failures and initiating events is constant with time and that the failure of any snubber on that system could cause the system to be unprotected and to result in failure during an assumed initiating event. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

The acceptance criteria are to be used in the visual inspection to determine OPERABILITY of the snubbers. For example, if a fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing.

To provide assurance of snubber functional reliability, one of three functional testing methods is used with the stated acceptance criteria:

Snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

Per Generic Letter 84-13

3/4.7.~~9~~ 8 SNUBBERS

~~All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.~~

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

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions, such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. ~~Selection of a representative sample according to the expression~~ $\cancel{35 (1 + \frac{c}{2})}$ ~~provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits~~. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

~~SNUBBERS (Continued)~~

~~1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or~~

~~2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1, or~~

~~3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation~~.

~~Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan~~.

~~Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions~~.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.~~10~~ 9 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

~~3/4.7.11 FIRE SUPPRESSION SYSTEMS~~

~~The OPERABILITY of the Fire Suppression Systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The Fire Suppression System consists of the water system, spray, and/or sprinklers, CO_2, Halon, fire hose stations, and yard fire hydrants.~~

~~The collective capability of the Fire Suppression Systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility Fire Protection Program.~~

~~In the event that portions of the Fire Suppression Systems are inoperable, alternate backup fire-fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire-fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.~~

~~The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the Fire Suppression Systems are met. An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.~~

~~In the event the Fire Suppression Water System becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.~~

~~3/4.7.12 FIRE RATED ASSEMBLIES~~

~~The functional integrity of the fire rated assemblies and barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. These design features minimize the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility Fire Protection Program and are subject to periodic inspections.~~

~~Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.~~

~~During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established until the barrier is restored to functional status.~~

3/4.7.~~13~~ 10 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ± ____°F.

See insert to page B 3/4 7-7.

Insert to B 3/4 7-7

3/4.7.11 ENGINEERED SAFETY FEATURES (ESF) ROOM COOLER AND SAFETY-RELATED CHILLER SYSTEM

The operation of the ESF Room Cooler and Safety-Related Chiller System ensures that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment cooled by the system.

3/4.7.12 REACTOR COOLANT PUMP THERMAL BARRIER COOLING WATER ISOLATION

This isolation function is designed to prevent a spill of the reactor coolant from a postulated breached thermal barrier should a break occur in the nonsafety-related ACCW piping downstream of the isolation valve.

3/4.8.1, 3/4.8.2, and 3/4.8.3 A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

Appendix A to

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974 and Generic Letter 84-15, "Proposed Staff Position to Improve and Maintain Diesel Generator Reliability." When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss-of-offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term, verify, as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the Surveillance Requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

based

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are ~~in accordance with~~ the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, Generic Letter 84-15, and Generic Letter 83-26, "Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests."

Appendix A to

A.C. SOURCES, D.C. SOURCES, and ONSITE POWER DISTRIBUTION (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," and 484-1975 "Recommended Practice for installation Design and Installation of Lead Storage Batteries for Generating Stations and Substation.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than ~~2.07~~ 2.10 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers ~~and fuses~~ provide assurance of breaker ~~and fuse~~ reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. ~~and/or fuse.~~ Each manufacturer's molded case and metal case circuit breakers ~~and/or fuses~~ are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers ~~and/or fuses~~ are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, ~~and/or fuses~~, it is necessary to divide that manufacturer's breakers ~~and/or fuses~~ into groups and treat each group as a separate type of breaker ~~or fuses~~ for surveillance purposes.

The ~~OPERABILITY [or]~~ [bypassing] except during startup and periodic testing of the motor-operated valves thermal overload protection [continuously] ~~[or] [during accident conditions] [by integral bypass devices]~~ ensures that the thermal overload protection ~~[during accident conditions]~~ will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the ~~[OPERABILITY] [or]~~ [bypassing] of the thermal overload protection [continuously] ~~[and] [or] [during accident conditions]~~ are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portions of the Reactor Coolant System. This action prevents flow to the RCS of unborated water by closing flowpaths from sources of unborated water. These limitations are consistent with the initial conditions assumed for the Boron Dilution Accident in the safety analyses.

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. ~~These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses~~. The value of 0.95 or less for K_{eff} includes a 1% Δk/k conservative allowance for uncertainties. Similarly, the boron concentration value of [2000] ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. The locking closed of the required valves during refueling operations precludes the possibility of uncontrolled boron dilution of the filled portion of the RCS. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the Source Range Neutron Flux Monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 ~~MANIPULATOR CRANE~~ REFUELING MACHINE

Rod Cluster Control Assemblies (RCCA)

The OPERABILITY requirements for the ~~manipulator cranes~~ refueling machine ensure that: (1) ~~manipulator cranes~~ Refueling Machine will be used for movement of ~~drive rods~~ and fuel assemblies, ~~(2) each crane~~ has sufficient load capacity to lift a ~~drive rod~~ RCCA or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) ~~loop~~ train be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR ~~loops~~ trains OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR ~~loop~~ train will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR ~~loop~~ train, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT ~~PURGE AND EXHAUST~~ VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

~~3/4.9.12 STORAGE POOL VENTILATION SYSTEM~~

~~The limitations on the Storage Pool Ventilation System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 10 continuous hours in a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1975 will be used as a procedural guide for surveillance testing.~~

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to: (1) measure control rod worth, and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 ~~which in turn may cause~~ and the RCS T_{avg} ~~to~~ may fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain STARTUP and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

This special test exception permits the Position Indication Systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time are derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, cannot be observed if the Position Indication Systems remain OPERABLE.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within: (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of

DOSE (Continued)

Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50 for liquid effluents.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Treatment Systems, the liquid effluents from the shared system are to be proportioned among the units sharing that system.

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks listed in this specification include all those outdoor radwaste tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section I.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established

DOSE-NOBLE GASES (Continued)

in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, "Revision I, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131 Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

GASEOUS WASTE PROCESSING

The OPERABILITY of the ~~WASTE GAS HOLDUP~~ SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

GASEOUS WASTE PROCESSING

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the ~~WASTE GAS HOLDUP~~ SYSTEM is maintained below the flammability limits of hydrogen and oxygen. [Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.] Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

DECAY

3/4 11.2.6 GAS ~~STORAGE~~ TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each gas ~~storage~~ tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem.

decay

This is consistent with Standard Review Plan 11.3, Branch Technical Position ETSB 11-5, "Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure," in NUREG-0800, July 1981.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to, waste type, waste pH, waste/liquid/SOLIDIFICATION agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

that directly support the production of electrical power for public use

This specification is provided to meet the dose limitations of 10 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units (including outside storage tanks, etc.) are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER of the PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

that may be compared with those

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring measuring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the Radiological Effluent Monitoring Program by ~~verifying that the measurable~~ concentrations of radioactive materials and levels of radiation ~~are not higher than~~ expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring, Revision 1, November 1979. ~~The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.~~

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L.A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), and in the HASL Procedures Manual, HASL-300 (revised annually).

3/4.12.2 LAND USE ~~CENSUS~~ SURVEY

This specification survey is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program are made if required by the results of this ~~census~~. ~~The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used~~. This ~~census~~ satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the ~~census~~ survey to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTIONS 3.0 AND 4.0 BASES

General:

These bases have been revised to be consistent with the Technical Specifications as marked in this draft. In addition, certain revisions have been made for the purposes of amplification and clarification.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure ~~[~~5.1-1~~]~~.

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure ~~[~~5.1-2~~]~~.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure ~~[~~5.1-~~3]~~ 1.

The definition of UNRESTRICTED AREA used in implementing these Technical Specifications has been expanded over that in 10 CFR 20.3(a)(17). The UNRESTRICTED AREA boundary may coincide with the Exclusion (fenced) Area boundary, as defined in 10 CFR 100.3(a), but the UNRESTRICTED AREA does not include areas over water bodies. The concept of UNRESTRICTED AREAS, established at or beyond the SITE BOUNDARY, is utilized in the Limiting Conditions for Operation to keep levels of radioactive materials in liquid and gaseous effluents as low as is reasonably achievable, pursuant to 10 CFR 50.36a. *The Site Boundary lines, plant property lines and the exclusion area lines are all the same. The property lines as shown in Figure 5.1-1 are the boundaries for determining effluent release limits.*

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment building is a steel-lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 226 feet.

c. Minimum thickness of concrete walls = 3 feet, *9 inches.*

d. Minimum thickness of concrete roof = 3 feet, *9 inches.*

e. Minimum thickness of concrete ~~floor pad~~ *basemat* = 9 feet, *3 inches.*

f. Minimum thickness of concrete instrumentation cavity = 8 feet.

g. ~~f.~~ Nominal thickness of steel liner = 1/4 inches.

h. ~~g.~~ Net free volume = 2.75×10^6 cubic feet.

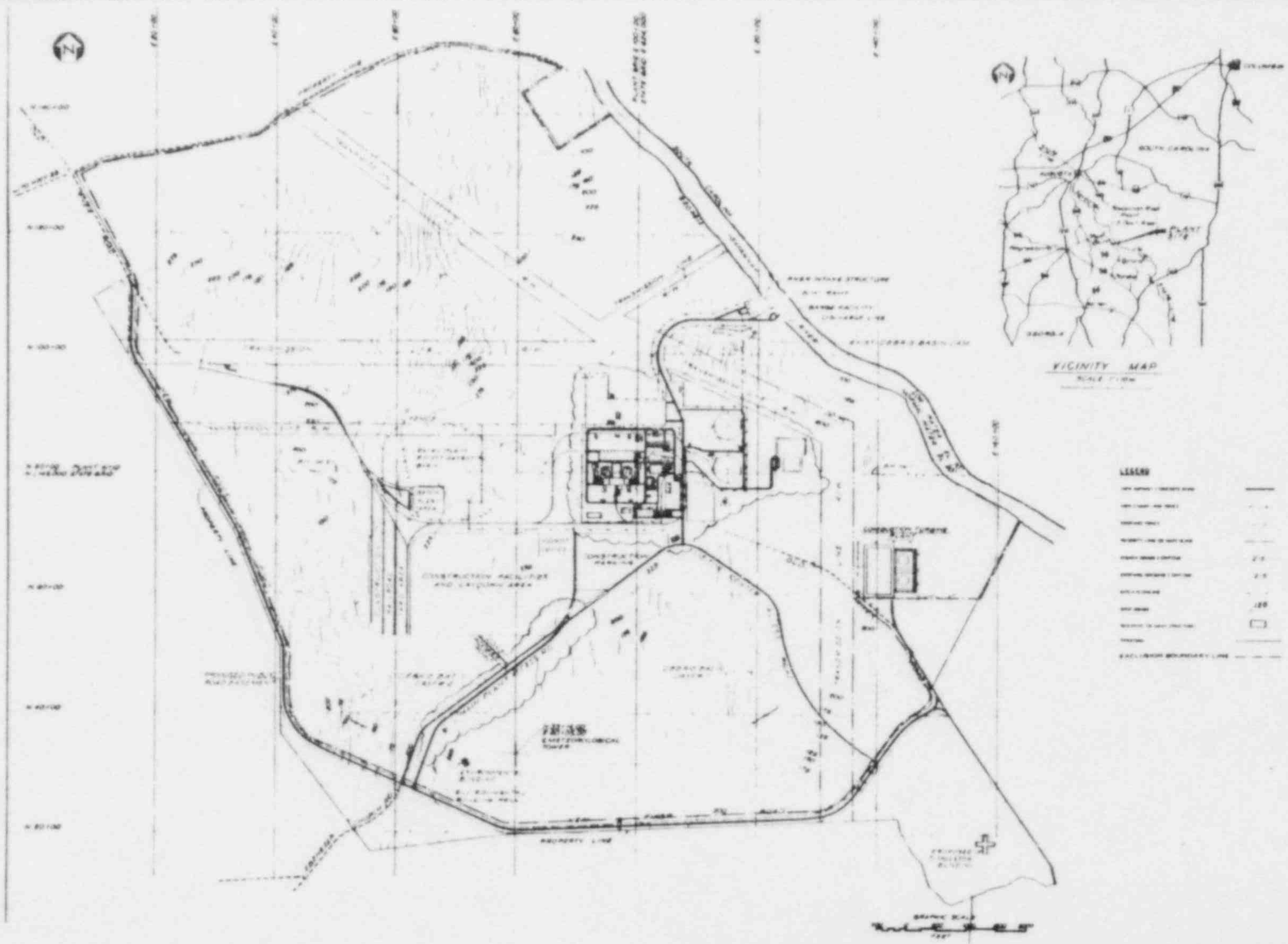
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 52 psig and a temperature of 400°F.

This figure shall consist of a map of the site area and provide at a minimum, the information described in Section [2.1.2] of the FSAR and meteorological tower location.

FIGURE 5.1-1

EXCLUSION AREA



CX2D46V002 REV 6

Georgia Power

VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

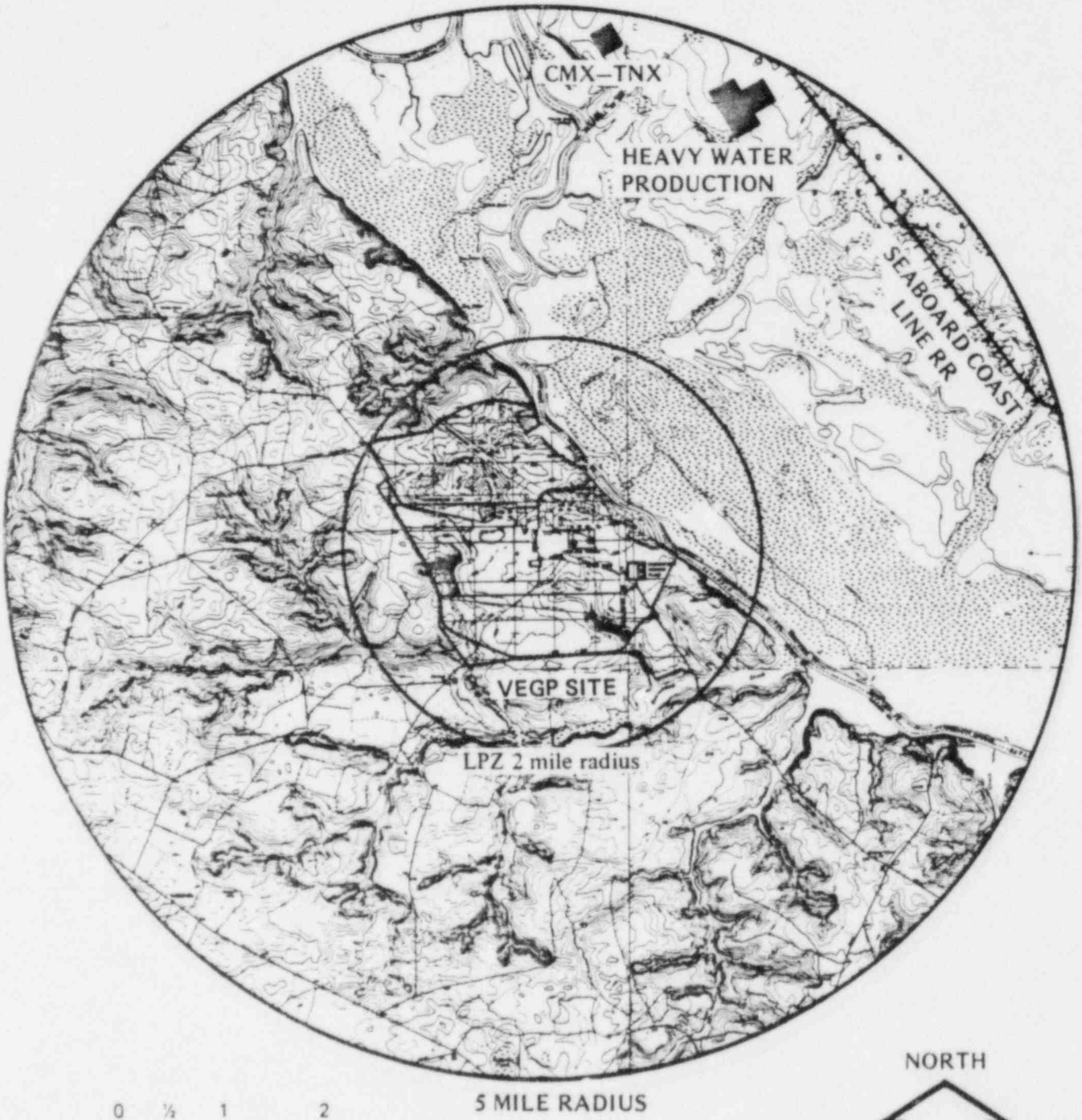
EXCLUSION AREA

FIGURE 5.1-1

This figure shall consist of a map of the site area showing the Low Population Zone boundary. Features such as towns, roads, industrial areas and recreational areas shall be indicated in sufficient detail to allow identification of significant shifts in population distribution within the LPZ.

FIGURE 5.1-2

LOW POPULATION ZONE



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VOGTLE
ELECTRIC GENERATING PLANT
UNIT 1 AND UNIT 2

~~AREA MAP~~
LOW POPULATION ZONE

FIGURE ~~2.1.1-3~~ 5.1-2

This figure shall consist of a map of the site area showing the SITE BOUNDARY and locating points within the SITE BOUNDARY where radioactive gaseous and liquid effluents are released, as well as where radioactive liquid effluents leave the site. If onsite areas subject to radioactive materials in gaseous or liquid effluents are utilized by the public for recreational or other purposes, these areas shall be outlined on the map and identified by occupancy factors and the licensee's method of occupancy control (if any). The figure shall be sufficiently detailed to allow identification of structures and release point locations and elevations, as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible by MEMBERS OF THE PUBLIC. The map scale shall be on the order of 2-3"/mile. See NUREG-0133 for additional guidance.

FIGURE 5.1-3

UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 193 fuel assemblies with each fuel assembly containing 264 fuel rods clad with ~~[~~Zircaloy-4~~]~~. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.5 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 53 ~~full-length~~ control rod assemblies. ~~The full-length~~ All control rod assemblies shall contain a nominal 142 inches of absorber material. ~~The nominal values of absorber material shall be ____% silver, ____% indium, and ____% cadmium.~~ The composition shall be 95.5% natural halfnium and 4.5% natural zirconium. All control rods shall be clad with stainless steel ~~tubing~~.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section ~~[~~5.2~~]~~ of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650 °F, except for the pressurizer which is 680 °F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 12,240 ± 100 cubic feet at a nominal T_{avg} of ~~[525]~~ 578.5 °F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure ~~[~~5.1-1~~]~~.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of ~~[2.6]% Δk/k~~ for uncertainties as described in Section {4.3} of the FSAR, and

b. A nominal ~~[21]~~ 10.6 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed {0.98} when aqueous foam moderation is assumed.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194'-1½".

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 288 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 ~~[250]~~ heatup cycles at $\leq$ 100°F/h and ~~[250]~~ 200 cooldown cycles at $\leq$ 100°F/h.	Heatup cycle - T_{avg} from $\leq$ 200°F to $\geq$ 550°F. Cooldown cycle - T_{avg} from $\geq$ 550°F to $\leq$ 200°F.
	200 ~~[250]~~ pressurizer cooldown cycles at $\leq$ 200°F/h.	Pressurizer cooldown cycle temperatures from $\geq$ 650°F to $\leq$ 200°F.
	80 ~~[100]~~ loss of load cycles, without immediate Turbine or Reactor trip.	$\geq$ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 ~~[50]~~ cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 ~~[100]~~ cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 ~~[500]~~ Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	[10] auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	200 ~~[50]~~ leak tests.	Pressurized to $\geq$ [2485] psig.
	10 ~~[5]~~ hydrostatic pressure tests.	Pressurized to $\geq$ 3107 ~~[3100]~~ psig.
Secondary Coolant System	[1] steam line break.	Break in a > 6-inch steam line.
	10 ~~[5]~~ hydrostatic pressure tests.	Pressurized to $\geq$ 1481 ~~[1350]~~ psig.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 5.0

5.1.3:

This specification was revised to reflect the fact that the site boundary lines, plant property lines, and the exclusion area are all the same. See Figure 1.1-1 of the VEGP FSAR.

5.3.2:

This section was revised to reflect plant-specific RCCA design. Reference to full-length rods was deleted as redundant since VEGP does not utilize part-length rods.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The ~~[Plant Superintendent]~~ General Manager-Vogtle Nuclear Operations (GMVNO) shall be responsible for overall ~~unit~~ plant operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 ~~The Shift Supervisor (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the [highest level of corporate management] shall be reissued to all station personnel on an annual basis.~~ The GMVNO will annually reissue a directive that emphasizes the primary management responsibility of the onshift operations supervisor (or during his absence from the control room, the individual designated to assume the command functions) for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for ~~unit~~ plant management and technical support shall be as shown in Figure 6.2-1.

~~UNIT STAFF~~ ONSITE

6.2.2 The ~~unit~~ onsite organization shall be as shown in Figure 6.2-2 and:

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;

b. At least one either licensed Operator shall be in the either reactor control room when fuel is in ~~the~~ reactor. In addition, while ~~the unit~~ is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

c. ~~A Health Physics Technician~~ An individual* qualified to implement radiation protection procedures shall be on site when fuel is in the reactor;

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who is in the containment building of the affected reactor and has no other concurrent responsibilities during this operation;

~~e. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and the [two] other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency; and~~

---

*The ~~Health Physics Technician~~ individual qualified to implement radiation protection procedures ~~and Fire Brigade composition~~ may be ~~less than the minimum requirements~~ absent for a period of time not to exceed ~~2~~ 4 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

ONSITE ~~UNIT~~ STAFF (Continued)

e. ~~f.~~ Administrative procedures shall be developed in performance of and implemented to limit the working hours of plant ~~unit~~ staff ~~who perform~~ safety-related functions (e.g., licensed Senior Operators, licensed Operators, Key Health Physics Technicians ~~health physicists~~, Key non-licensed ~~auxiliary~~ operators, and key maintenance personnel).

~~[The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter No. 82-12).]~~

or

[Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating plant personnel work a ~~normal 8-hour day,~~ 40-hour week while the plant ~~unit~~ is operating. (this work week may consist of 12-hour shift schedules) However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the applicable Department Superintendent ~~[Plant Superintendent] or his deputy~~, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual excess overtime shall be reviewed monthly by the General Manager - Vogtle Nuclear Operations ~~[Plant Superintendent]~~ or his designee to assure that excessive hours were authorized and that they do not become routine. ~~have not been assigned. Routine deviation from the above guidelines is not authorized.]~~

This figure shall show the organizational structure and lines of responsibility for the offsite groups that provide technical and management support for the unit. The organizational arrangement for performing and monitoring quality assurance activities shall also be indicated.

FIGURE 6.2-1

OFFSITE ORGANIZATION

This figure shall show the organizational structure and lines of responsibility for the unit staff. Positions to be staffed by licensed personnel shall be indicated. The organizational arrangement for performing and monitoring quality assurance activities shall also be indicated.

FIGURE 6.2-2

UNIT ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SINGLE UNIT FACILITY

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1*	None

SS - Shift Supervisor with a Senior Operator license on Unit 1
SRO - Individual with a Senior Operator license on Unit 1
RO - Individual with an Operator license on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

*The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

See insert to page 6-5

~~TABLE 6.2-1a~~

~~MINIMUM SHIFT CREW COMPOSITION~~

~~TWO UNITS WITH A COMMON CONTROL ROOM~~

~~POSITION~~	~~NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION~~		
	~~BOTH UNITS IN MODE 1, 2, 3, or 4~~	~~BOTH UNITS IN MODE 5 or 6 OR DEFUELED~~	~~ONE UNIT IN MODE 1, 2, 3 or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED~~
~~SS~~	~~1~~	~~1~~	~~1~~
~~SRO~~	~~1~~	~~none**~~	~~1~~
~~RO~~	~~3*~~	~~2*~~	~~3*~~
~~AO~~	~~3*~~	~~3*~~	~~3*~~
~~STA~~	~~1***~~	~~none~~	~~1***~~

OS ~~SS~~ - ~~Shift~~ Operations Supervisor with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
NLO ~~AO - Auxiliary Operator~~ Non-Licensed Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed ~~2~~ 4 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the ~~Shift~~ Operations Supervisor from the control room while ~~the unit~~ either reactor is in MODE 1, 2, 3, or 4, an individual ~~(other than the Shift Technical Advisor)~~ with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the ~~Shift~~ Operations Supervisor from the control room while ~~the unit~~ either reactor is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* At least one of the required individuals must be assigned to the designated position for each unit.

** At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

*** The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

Insert to Page 6-5.

TABLE 6.2-1
MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH A COMMON CONTROL ROOM

UNIT 1 IN MODES 1, 2, 3, OR 4

	UNIT 2 IN MODE:		
POSITION	1, 2, 3, OR 4	5 OR 6	DEFUELED
OS	1	1	1
SRO	1	1	1
RO	3	3	2
NLO	3	3	2
STA*	1	1	1

UNIT 1 IN MODE 5 OR 6

	UNIT 2 IN MODE:		
POSITION	1, 2, 3, OR 4	5 OR 6	DEFUELED
OS	1	1	1
SRO	1	0	0
RO	3	2	1
NLO	3	3	1
STA*	1	0	0

UNIT 1 DEFUELED

	UNIT 2 IN MODE:		
POSITION	1, 2, 3, OR 4	5 OR 6	DEFUELED
OS	1	1	0
SRO	1	0	0
RO	2	1	0
NLO	2	1	0
STA*	1	0	0

* THE STA IS NOT REQUIRED ON SHIFT IF THE QUALIFICATIONS OF THE OPERATIONS SUPERVISOR, ARE UPGRADED TO OR SRO FULFILL THE REQUIREMENTS OF THE STA POSITION.

TABLE 6.2-1b

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH TWO SEPARATE CONTROL ROOMS

WITH UNIT [2] IN MODE 5 OR 6 OR DEFUELED

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1*	1*
SRO	1	None
RO	2	1
AO	2	2**
STA	1***	None

WITH UNIT [2] IN MODE 1, 2, 3, OR 4

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS	1*	1*
SRO	1	None
RO	2	1
AO	2	1
STA	1* ***	None

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Supervisor from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

* Individual may fill the same position on Unit [2].

** One of the two required individuals may fill the same position on Unit [2].

*** The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Supervisor the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant ~~unit~~ operating characteristics, NRC issuances, industry advisories, ~~Licensee Event Reports,~~ and other sources of plant ~~unit~~ design and operating experience information, ~~including units of similar design~~, which may indicate areas for improving plant ~~unit~~ safety. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant ~~unit~~ safety to ~~[a high level corporate official in a technically oriented position who is not in the management chain for power production]~~. the Senior Vice President-Nuclear Operations through the Manager-Nuclear Performance and analysis.

COMPOSITION

6.2.3.2 The ISEG shall be composed of at least five, dedicated, full-time engineers. ~~located on site~~. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant operations and maintenance ~~unit~~ activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

RECORDS

6.2.3.4 Records of activities performed by the ISEG shall be prepared, maintained, and forwarded each calendar month to [a high level corporate official in a technically oriented position who is not in the management chain for power production].

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room.

~~6.3 UNIT STAFF QUALIFICATIONS~~

~~[Minimum qualifications for members of the unit staff shall be specified by use of an overall qualification statement referencing an ANSI Standard acceptable to the NRC staff or alternately by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of a unique organizational structure.]~~

*Not responsible for sign-off function.

Replace with Insert A to page 6-7.

6.3 PLANT STAFF QUALIFICATIONS
~~UNIT STAFF QUALIFICATIONS (Continued)~~

~~6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of [an ANSI Standard acceptable to the NRC staff] for comparable positions, except for the [Radiation Protection Manager] who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees~~.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the ~~unit~~ plant staff shall be maintained under the direction of the ~~[position title] and shall meet or exceed the requirements and recommendations of Section ____ of [an ANSI Standard acceptable to the NRC staff] and Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience~~. Superintendent of Nuclear Training

Replace with Insert B to page 6-7.

6.5 REVIEW AND AUDIT

~~[The method by which independent review and audit of unit operations is accomplished may take one of several forms. The licensee may either assign this function to an organizational unit separate and independent from the group having responsibility for unit operation or may utilize a standing committee composed of individuals from within and outside the licensee's organization~~.

~~Irrespective of the method used, the licensee shall specify the details of each functional element provided for the independent review and audit process as illustrated in the following example specifications.]~~

~~6.5.1 [UNIT REVIEW GROUP (URG)]~~ PLANT REVIEW BOARD (PRB)

FUNCTION

6.5.1.1 The ~~[URG]~~ PRB shall function to advise the ~~[Plant Superintendent]~~ GMVNO on ~~all~~ matters related to nuclear safety.

COMPOSITION

~~6.5.1.2 The [URG] shall be composed of the~~:

Replace with Insert C to page 6-7.

~~Chairman:~~	~~[Plant Superintendent]~~
~~Member:~~	~~[Operations Supervisor]~~
~~Member:~~	~~[Technical Supervisor]~~
~~Member:~~	~~[Maintenance Supervisor]~~
~~Member:~~	~~[Plant Instrument and Control Engineer]~~
~~Member:~~	~~[Plant Nuclear Engineer]~~
~~Member:~~	~~[Health Physicist]~~

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the ~~[URG]~~ PRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ~~[URG]~~ PRB activities at any one time.

Insert A to 6-7

Personnel will meet the minimum education and experience recommendations of ANSI/ANS N18.1-1971 before they are considered qualified to perform all duties independently. Prior to meeting the recommendations of ANSI N18.1-1971, personnel may be trained to perform specific tasks and will be qualified to perform those tasks independently.

Insert B to 6-7

Personnel will meet the minimum education and experience recommendations of ANSI/ANS N18.1-1971 and, for licensed staff, Appendix A of 10CFR55 before they are considered qualified to perform all duties independently. Prior to meeting the recommendations of ANSI/ANS N18.1-1971, personnel may be trained to perform specific tasks and will be qualified to perform those tasks independently.

Insert C to 6-7

6.5.1.2 The PRB shall be composed of, as a minimum, a supervisor or equally qualified individual from the departments listed below:

Operations
Maintenance
Quality Control (QC)
Health Physics
Regulatory Compliance
Plant Engineering and Services

The Chairman and his alternate and other members of the PRB shall be designated by the GMVNO.

MEETING FREQUENCY

6.5.1.4 The ~~[URG]~~ PRB shall meet at least once per calendar month and as convened by the ~~[URG]~~ PRB Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the ~~[URG]~~ PRB necessary for the performance of the ~~(URG)~~ PRB responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

see insert to pages 6-8 and 6-9 for items a and b.

6.5.1.6 The ~~(URG)~~ PRB shall be responsible for:

~~a. Review of: (1) all proposed procedures required by Specification 6.8 and changes thereto, (2) all proposed programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the [Plant Superintendent] to affect nuclear safety;~~

c. ~~b.~~ Review of all proposed tests and experiments that affect nuclear safety;

d. ~~c.~~ Review of all proposed changes to the ~~Appendix "A"~~ Technical Specifications;

e. ~~d.~~ Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

f. ~~e.~~ 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]~~ Vice President and General Manager - Nuclear Operations and to the ~~[Company Nuclear Review and Audit Group]~~ Safety Review Board;

g. ~~f.~~ Review of all REPORTABLE EVENTS;

h. ~~g.~~ Review of evaluations of plant ~~unit~~ operations to detect potential hazards to nuclear safety;

i. ~~h.~~ Performance of special reviews, investigations, or analyses and reports thereon as requested by the ~~[Plant Superintendent]~~ GMVNO or the ~~[Company Nuclear Review and Audit Group]~~ Safety Review Board;

j. ~~i.~~ Review of the Security Plan and implementing procedures and submittal of recommended changes to the ~~[Company Nuclear Review and Audit Group]~~ GMVNO and the Safety Review Board;

RESPONSIBILITIES (Continued)

k. ~~j.~~ Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the ~~[Company Nuclear Review and Audit Group]~~ GMVNO and the Safety Review Board;

See insert to pages 6-8 and 6-9 for item l and item n.

l. ~~k. Review of any accidental, unplanned, or uncontrolled radioactive release 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 [Company Nuclear Review and Audit Group]; and~~

m. ~~l.~~ Review of changes to the PROCESS CONTROL PROGRAM, the OFFSITE DOSE CALCULATION MANUAL, and the Radwaste Treatment Systems.

6.5.1.7 The ~~[URG]~~ PRB shall:

a. Recommend in writing to the ~~[Plant Superintendent]~~ GMVNO approval or disapproval of items considered under Specification 6.5.1.6a. through ~~d.~~ e above prior to their implementation;

b. Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through ~~e.~~ f constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the ~~[Vice President-Nuclear Operations]~~ Vice President and General Manager-Nuclear Operations and the ~~[Company Nuclear Review and Audit Group]~~ Safety Review Board of disagreement between the ~~[URG]~~ PRB and the ~~[Plant Superintendent]~~ GMVNO; however, the ~~[Plant Superintendent]~~ GMVNO shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1.

RECORDS

6.5.1.8 The ~~[URG]~~ PRB shall maintain written PRB minutes of each ~~[URG]~~ PRB meeting that, at a minimum, document the results of ~~all [URG]~~ activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the [Vice President-Nuclear Operations] and the ~~[Company Nuclear Review and Audit Group]~~ Safety Review Board.

6.5.2 ~~[COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNRAG)]~~ SAFETY REVIEW BOARD (SRB)

FUNCTION

6.5.2.1 The ~~[CNRAG]~~ SRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

INSERT FOR Pages 6-8 and 6-9

a. Review of 1) procedures which establish plant-wide administrative controls to implement the Q.A. program or Technical Specification surveillance program, 2) procedures for changing plant operating modes, 3) emergency and abnormal operating procedures, 4) procedures for effluent releases of radiological consequence, 5) fuel handling procedures.

b. Review of 1) programs required by Specification 6.8.4 and changes thereto, 2) proposed procedures and changes to procedures, which involve an unreviewed safety question as per 10 CFR 50.59.

l. Review of any accidental, unplanned, or uncontrolled radioactive release in excess of 1 Ci, excluding dissolved and entrained gasses and tritium for liquid effluents, and in excess of 150 Ci for noble gasses or 0.02 Ci of radioiodines for gaseous effluents; and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President and General Manager-Nuclear Operations and the Safety Review Board.

n. Review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the GMVNO.

FUNCTION (Continued)

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering,

h. Quality assurance practices, and

~~i. [Other appropriate fields associated with the unique characteristics of the nuclear power plant.]~~

The ~~[CNRAG]~~ SRB shall report to and advise the senior [Vice President-Nuclear Operations] on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION

6.5.2.2 The ~~[CNRAG]~~ shall be composed of ~~the~~: a minimum of five persons who, as a group, provide the expertise to review and audit the operation of a nuclear power plant. The Chairman and other members shall be appointed by the Senior Vice President - Nuclear Operations or such other person as he shall designate. The composition of the SRB shall meet the requirements of ANSI N 18.7-1976

~~Director: [Position Title]~~
~~Member: [Position Title]~~
~~Member: [Position Title]~~
~~Member: [Position Title]~~
~~Member: [Position Title]~~

ALTERNATES

6.5.2.3 All alternate ~~members~~ shall be appointed in writing by the ~~[CNRAG]~~ SRB ~~Director~~ Chairman to serve on a temporary basis; however, no more than ~~two~~ a minority of alternates shall participate as voting members in ~~[CNRAG]~~ SRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the ~~(CNRAG) Director~~ SRB Chairman to provide expert advice to the ~~[CNRAG]~~ SRB.

MEETING FREQUENCY

6.5.2.5 The ~~[CNRAG]~~ SRB shall meet at least once per calendar quarter during the initial year of ~~unit~~ plant operation following fuel loading and at least once per 6 months thereafter.

Insert to page 6-10 here.

Insert to page 6-10

However, in extenuating circumstances, the senior Vice President-Nuclear Operations may designate the use of additional alternates with voting authority when regular members are not available within necessary time constraints.

QUORUM

6.5.2.6 The quorum of the SRB ~~[CNRAG]~~ necessary for the performance of the SRB ~~[CNRAG]~~ review and audit functions of these Technical Specifications shall consist of the Chairman ~~Director~~ or his designated alternate and at least ~~[four CNRAG]~~ members including alternates. No more than a minority of the quorum a majority of the shall have line responsibility for operation of the plant ~~unit~~.

REVIEW

6.5.2.7 The SRB ~~[CNRAG]~~ shall be responsible for the review of:

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question;

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

d. Proposed changes to Technical Specifications or this Operating License;

e. Violations of Codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance;

f. Significant operating abnormalities or deviations from normal and expected performance of plant ~~unit~~ equipment that affect nuclear safety;

g. All REPORTABLE EVENTS;

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety; and

i. Reports and meeting minutes of the PRB ~~[URG]~~.

AUDITS

6.5.2.8 Audits of plant ~~unit~~ activities shall be performed under the cognizance of the SRB ~~[CNRAG]. These audits shall encompass~~: See insert for page 6-11.

a. The conformance of plant ~~unit~~ operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

INSERT for Page 6-11

Each inspection or audit shall be performed within the specified time interval with:

1. A maximum allowable extension not to exceed 25% of the inspection audit interval.

2. A total maximum combined interval time for any 3 consecutive inspection or audit intervals not to exceed 3.25 times the specified inspection or audit interval.

These audits shall encompass:

AUDITS (Continued)

b. The performance, training, and qualifications of the entire plant ~~unit~~ staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in plant ~~unit~~ equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

~~k. Any other area of unit operation considered appropriate by the [CNRAG] or the [Vice President-Nuclear Operations].~~

See insert for page 6-12.

RECORDS

6.5.2.9 Records of SRB ~~[CNRAG]~~ activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of ~~each~~ SRB ~~[CNRAG]~~ meeting shall be prepared, approved, and forwarded to the Senior ~~[~~Vice President-Nuclear Operations~~]~~ within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Senior ~~[~~Vice President-Nuclear Operations~~]~~ within 14 days following completion of the review; and

Insert for page 6-12

l. The Emergency Plan and implementing procedures (at least once per 12 months); and

m. The Security Plan and implementing procedures (at least once per 12 months).

RECORDS (Continued)

Executive Vice President - Power Supply,

Senior

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the ~~[Vice President-Nuclear Operations]~~ and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Section 50.72 and Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the ~~[URG]~~ PRB, and the results of this review shall be submitted to the ~~[CNRAG]~~ SRB and the ~~[Vice President-Nuclear Operations]~~.

Vice President and General Manager - Nuclear Operations

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. ~~The Manager, Nuclear Production and the NFSC~~ shall be notified within 24 hours.

The Vice President and General Manager - Nuclear Operations, the SRB, PRB, and the GMVNO

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The Licensee Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the ~~PORC~~ PRB, SRB, the ~~NFSC~~ GMVNO, and the ~~Manager, Nuclear Production~~ Vice President and General Manager - Nuclear Operations within 30 days after discovery of the event.

d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

6.8 PROCEDURES AND PROGRAMS

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

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

PROCEDURES AND PROGRAMS (Continued)

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

See insert for page 6-14.

g. Quality Assurance for effluent and environmental monitoring.

h. Fire Protection Program implementation.

6.8.2 ~~Each procedure of Specification 6.8.1, and changes thereto, shall be reviewed by the [URG] and shall be approved by the [Plant Superintendent] prior to implementation and reviewed periodically as set forth in administrative procedures~~.

6.8.3 Temporary changes to procedures of Specification 6.8.1 may be made provided:

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

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license. ~~on the unit affected; and~~

c. The change is documented, reviewed ~~by the [URG]~~, and approved, in accordance with 6.8.2 ~~by the [Plant Superintendent]~~ within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include [the, applicable portions of containment recirculation spray, Safety Injection, chemical and volume control, ~~gas stripper, and hydrogen recombiners~~, and residual heat removal systems]. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

2) ~~Integrated~~ leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

Insert for Page 6-14

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be approved by either the GMVNO or the department head of the responsible department prior to implementation with the exception of the following which shall be approved by the GMVNO:

1) procedures which establish plant-wide administrative controls (which implement the quality assurance program and the Technical Specifications surveillance program),

2) unit operating procedures (UOPs)

3) emergency operating procedures (EOPs)

4) abnormal operating procedures (AOPs)

5) procedures for implementing the security plan, emergency plan, and the fire protection program, and

6) fuel handling procedures.

PRB responsibilities for review of procedures are delineated in 6.5.1.6. Additionally, procedures will be reviewed periodically as set forth in administrative procedures.

PROCEDURES AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring (Continued)

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, ~~which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage~~,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

~~d. Backup Method for Determining Subcooling Margin~~
~~[PWRs with a single channel of monitoring instrumentation]~~

~~A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:~~

~~1) Training of personnel, and~~

~~2) Procedures for monitoring.~~

d ~~e.~~ Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

The initial Startup Report shall address each of the startup tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report. Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications.

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following 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 operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the ~~unit~~ plant as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of ~~station~~ plant, utility, and other personnel (including contractors) receiving exposures

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*A single submittal may be made for a multiple unit ~~station~~ plant. The submittal should combine those sections that are common to all ~~units~~ both reactors at the ~~station~~ plant.

ANNUAL REPORTS (Continued)

greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions* (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance [describe maintenance], waste processing, and refueling). The dose assignments to various duty functions may be estimated based on pocket dosimeter, thermoluminescent 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 should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration (μCi/gm) and one other radioidine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

See insert for page 6-17.

~~c. [Any other unit unique reports required on an annual basis.]~~

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE ~~OPERATING~~ REPORT**

6.9.1.3 Routine Annual Radiological Environmental Surveillance ~~Operating~~ Reports covering activities of the Radiological Environmental Monitoring Program ~~the operation of the unit~~ during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality and shall include copies of reports of the preoperational Radiological Environmental Monitoring Program of the unit for at least two years prior to initial criticality.

The Annual Radiological Environmental Surveillance ~~Operating~~ Reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period,

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*This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

**A single submittal may be made for a multiple unit ~~station~~ plant.

INSERT for Page 6-17

c. A report shall be prepared and submitted to the commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

d. An annual data report on diesel generator reliability will be submitted and, in addition, the following information will be included:

1. A summary of all tests (valid and invalid) that occurred within the time period over which the last 20/100 valid tests were performed.

2. Analysis of failures and determination of root ca. . of failures.

3. Identification of all actions taken or to be taken to 1) correct the root causes of failures defined in b) above and 2) achieve a general improvement of diesel generator reliability.

4. An assessment of the existing reliability of electric power to engineered-safety-feature equipment.

The radiological level of radionuclides which are not included in the plant effluents need not be reported.

ANNUAL RADIOLOGICAL ENVIRONMENTAL ~~OPERATING~~ SURVEILLANCE REPORT (Continued)

including, as appropriate, a comparison with preoperational studies, with operational controls, ~~as appropriate,~~ and with previous environmental surveillance reports, and an assessment of ~~the~~ any observed impacts of ~~the~~ plant operations on the environment. The reports shall also include the results of the Land Use ~~Census~~ Surveys required by Specification 3.12.2.

The Annual Radiological Environmental ~~Operating~~ Surveillance Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as ~~possible~~ practicable in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; ~~at least two~~ legible maps~~*~~ covering all sampling locations keyed to a table giving distances and directions from ~~the centerline of one~~ a point midway between the two reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not ~~achievable~~ achieved.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*~~*~~

6.9.1.4 Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data

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~~*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.~~

**A single submittal may be made for a multiple unit ~~station~~ plant. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., LSA, Type A, Type B, Large Quantity) and SOLIDIFICATION agent or absorbent (e.g., cement, urea formaldehyde).

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.* This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1 [~~5.1-3~~]) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM). Historical annual average meteorological conditions or

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other ~~nearby~~ uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1, October 1977.

within 8 Km

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid,

*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use ~~Census~~ Surveys pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.10 or 3.3.3.11, respectively; and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6, respectively.

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

RADIAL PEAKING FACTOR LIMIT REPORT

6.9.1.6 The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided to the NRC Regional Administrator with a copy to Director of Nuclear Reactor Regulation, Attention: Chief, Core Performance Branch, U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, for all core planes containing Bank "D" control rods and all unrodded core planes and the plot of predicted ($F_q^T \cdot P_{Rel}$) vs Axial Core Height with the limit envelope at least 60 days prior to each cycle initial criticality unless otherwise approved by the Commission by letter. In addition, in the event that the limit should change requiring a new substantial or an amended submittal to the Radial Peaking Factor Limit Report, it will be submitted 60 days prior to the date the limit would become effective unless otherwise approved by the Commission by letter. Any information needed to support F_{xy}^{RTP} will be by request from the NRC and need not be included in this report.

SPECIAL REPORTS

~~[Special reports may be required covering inspection, test, and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.]~~

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of ~~unit~~ reactor operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; ~~and~~

h. Records of annual physical inventory of all sealed source material of record; and

i. Records of secondary water sampling and water quality.

RECORD RETENTION (Continued)

6.10.3 The following records shall be retained for the duration of the ~~unit~~ plant Operating License:

a. Records and drawing changes reflecting ~~unit~~ plant design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of radiation exposure for all individuals entering radiation control areas;

d. Records of gaseous and liquid radioactive material released to the environs;

e. Records of transient or operational cycles for those ~~unit~~ plant components identified in Table 5.7-1;

f. Records of reactor tests and experiments;

g. Records of training and qualification for current members of the ~~unit~~ plant staff;

h. Records of inservice inspections performed pursuant to these Technical Specifications;

i. Records of quality assurance activities required by the ~~Operational Quality Assurance Manual~~ FSAR;

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

k. Records of meetings of the ~~[URG]~~ FRB and the ~~[CNRAG]~~ SRB;

l. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.~~8~~9 including the date at which the service life commences and associated installation and maintenance records;

~~m. Records of secondary water sampling and water quality; and~~

m. ~~n.~~ Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA [OPTIONAL]

greater than 100 mrem/hr but

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is ~~equal to or~~ less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) shall or personnel ~~continuously~~ escorted by such individuals ~~may~~ be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ~~equal to or~~ less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area; or

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the [Radiation Protection Manager] in the RWP.

See insert for page 6-23

6.12.2 ~~In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided~~ with locked doors to prevent unauthorized entry~~, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or~~ health physics supervision. ~~Doors shall remain locked except during periods of access~~ by personnel under an approved RWP ~~which shall specify the~~ dose rate levels in the immediate work areas and the ~~maximum allowable stay time for individuals in that area.~~ In lieu of the ~~stay time specification of the RWP~~, direct or remote (such as closed circuit TV cameras) ~~continuous surveillance~~ may be made by personnel ~~qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area~~.

~~For individual~~ high ~~radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment,~~ where no enclosure exists for ~~purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device~~.

INSERT FOR Page 6-23

The requirements of 6.12.1, above, shall also apply to those areas accessible to personnel with radiation levels such that a major portion of the body could receive a dose greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas. Those areas for which no enclosure exists for purposes of locking (and no enclosure can be reasonably constructed around those areas) shall be roped-off, conspicuously posted, and a flashing light shall be activated as a warning device. Dose measurements are made at 18" from the source of radioactivity in these areas. The keys shall be maintained under the administrative control of the Operations Supervisor on duty and/or the Plant Health Physicist.

6.13 PROCESS CONTROL PROGRAM (PCP)

~~6.13.1 The PCP shall be approved by the Commission prior to implementation.~~

6.13.~~2~~ 1 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the ~~[URG]~~ PRB.

b. Shall become effective upon review ~~and acceptance by the [URG].~~ in accordance with Specification 6.5.1.6 and approval in accordance with Specification 6.8.2.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

~~6.14.1 The ODCM shall be approved by the Commission prior to implementation.~~

6.14.~~2~~ 1 Licensee-initiated changes to the ODCM:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

3) Documentation of the fact that the change has been reviewed and found acceptable by the ~~[URG]~~ PRB.

b. Shall become effective upon review ~~and acceptance by the [URG].~~ in accordance with Specification 6.5.1.6 and approval in accordance with Specification 6.8.2.

6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~[URG]~~ PRB. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable ~~by the [URG].~~ in accordance with Specification 6.5.1.

b. Shall become effective upon review and acceptance ~~by the [URG].~~ in accordance with Specification 6.5.1.

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

6.16 AMENDMENTS TO TECHNICAL SPECIFICATIONS

The effective date of NRC approved amendments to the VEGP Technical Specifications shall be at least 60 days from the date of issuance of the amendment unless otherwise requested in writing from GPC.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 6.0

General:

"Plant Superintendent" was changed to "GMVNOD" to reflect plant-specific nomenclature.

The word "unit" was replaced by "plant" to reflect plant-specific nomenclature.

6.1.2:

The STS wording was replaced with the wording found in paragraph 13.5.1.1.E of the VEGP FSAR.

6.2.2:

All changes to 6.2.2 were made to reflect plant-specific nomenclature except the following:

1) "Health Physics Technician" was revised to "Individual qualified to implement Radiation Protection Procedures" on the basis that the intent of this specification can be achieved by the qualification of other individuals outside the Health Physics organization (i.e., any member of the operations organization-OS, [illegible], STA, RO, NLO) as long as the total number of onshift personnel is sufficient for the safe shutdown of the reactor(s) and the fire brigade.

2) The words "is in the containment building of the affected reactor and" were inserted to provide further definition of movement limitations necessary to meet the intent of "observed and directly supervised."

3) The deletion of 6.2.2.e was made on the basis that "personnel required for other essential functions during a fire emergency" is not well defined and the three members of the minimum shift crew will be able to achieve safe shutdown.

4) The time interval of 2 hours in footnote * was revised to 4 hours on the basis of plant location and personnel availability.

6.2.2.f:

The words "normal 8-hour day" were deleted and reference to a 12-hour shift schedule was added to be consistent with plant policy. The intent of Generic Letter 82-12 is maintained.

"Manager, Plant Operations" was revised to "Operations Superintendent" to be consistent with the plant-specific organization.

The words "or his deputy" were deleted to be consistent with plant policy.

The words "who perform" were replaced with "in performance of" in order to ensure that hours worked in performance of nonsafety-related functions are not included in the scope of this specification.

The term "health physicists" was replaced with "Key Health Physics Technicians" and "auxiliary operators" was replaced with "key non-licensed operators" in order to better define the word "Staff." In addition, certain health physics technicians and nonlicensed operators may perform work that is not readily subject to degraded quality due to extended hours.

The words "[Plant Superintendent] or his deputy" were replaced with "applicable department superintendent" to be consistent with paragraph 13.5.1.1.G of the VEGP FSAR. In addition, this paragraph was revised to clarify what the General Manager-Vogtle Nuclear Operations reviews with regard to excess overtime.

Although 12-hour shift schedules are in conflict with a "normal 8-hour day," these schedules meet the intent of the total number of hours worked during the 6-weeks rotating shift period. Fatigue should not be a factor as the 12-hour shift schedules allow for several consecutive days off during the 6-weeks rotating shift period.

Table 6.2-1:

The time interval for unexpected absences was revised from 2 hours to 4 hours on the basis of plant location with reference to personnel availability.

The format of Table 6.2-1 was revised for the sake of readability. The content is unchanged.

6.2.3:

The words "Licensee Event Report" were deleted from paragraph 6.2.3.1 to be consistent with item I.b.1.2 of NUREG-0737. Also, both the PRB and the SRB review Reportable Events.

The words "including units of similar design" were deleted on the basis that "NRC issuances, industry advisories, and other sources of plant design and operating experience information" will include any information regarding units of similar design. Also, the deleted phrase is not included in NUREG-0737, item I.B.1.2.

The words "located on site" were deleted since the five individuals will be a combination of corporate and onsite personnel. This is consistent with NUREG-0737, item I.B.1.2. (NUREG-0737, item I.B.1.2 provides for a smaller onsite group when a utility has multiple sites.)

6.3:

This statement is consistent with VEGP FSAR paragraph 13.1.3.1.

6.4.1:

This specification has been revised to be consistent with subsection 13.2.2 of the VEGP FSAR.

6.5.1.2:

This specification has been revised to be consistent with subsection 13.4.1 of the VEGP FSAR with the exception that we no longer have the title Vice-Chairman of the PRB. The chairman has an alternate who shall be designated by the GMVNO.

6.5.1.6:

a) & b) The VEGP PRB reviews the administrative and controls required to develop plant procedures. This should be adequate to ensure that acceptable procedures are generated. The VEGP PRB responsibilities are listed in subsection 13.4.1 of the VEGP FSAR. The revisions proposed here are consistent with paragraph 13.4.1.B of the FSAR.

d) "Appendix A" was deleted on the basis that the distinction between Appendix A and B is no longer warranted due to the incorporation of NUREG-0472 into NUREG-0452 and the redesignation of Appendix B as the "Environmental Protection Plan."

h) This specification was revised to be consistent with subsection 13.4.1 of the VEGP FSAR.

l) This specification was revised to be consistent with 13.4.1.k of the VEGP FSAR.

6.5.1.7:

a) The change "(a) through (d)" to "(a) through (e)" was made because of the change made to item 6.5.1.6.a.

b) The change "(a) through (e)" to "(a) through (f)" was made because of the change made to item 6.5.1.6.a.

The "Nuclear Review and Audit Group" will be known as the "Safety Review Board." These changes are specific to the organization of GPC and the VEGP.

6.5.2:

The Company Nuclear Review and Audit Group (CNRAG) will be known as the Safety Review Board (SRB). This revision has been made throughout the draft of Section 6.0.

6.5.2.2:

The wording supplied in this section provides a better description of the composition of the SRB than does the STS and is consistent with subsection 13.4.2 of the VEGP FSAR.

6.5.2.3 and 6.5.2.4:

The wording supplied in this section is consistent with subsection 13.4.2 of the VEGP FSAR.

6.5.2.6:

The wording supplied in this section is consistent with subsection 13.4.2 of the VEGP FSAR.

6.5.2.8:

The insert for page 6-11 was added to provide some flexibility for inspections and audits.

6.6.1:

Reference to Section 50.72 was added because it requires immediate notification and the word "or" was added because a written followup report may not be required.

6.7.1:

Addition of PRB and SRB is for clarification in that these bodies review Technical Specification violations.

6.8.2:

This section was revised to reflect the plant-specific method of review and approval of VEGP procedures and changes thereto.

6.8.3:

b) VEGP Units 1 and 2 will be essentially identical and the operations personnel licensed on Unit 1 will be licensed on Unit 2 also.

c) See the discussion for 6.8.2.

6.8.4:

a) The system designations were revised to be consistent with plant-specific terminology.

The word "integrated" was deleted to eliminate any confusion of this test with the containment ILRT requirements.

c) (3) At VEGP, monitoring can be performed prior to discharge. This is an accurate method for determining evidence of condenser in-leakage.

Old item d) This item was deleted on the basis that VEGP is equipped with more than one channel of monitoring instrumentation.

6.9:

See justifications for revisions to Specs. 3.4.8, 4.7.9.3, and 4.8.1.1.3.

6.9.1.3:

This annual report which documents radiological environmental monitoring activities for the calendar year might more appropriately be entitled the Annual Radiological Environmental Surveillance Report; this title is used at our other nuclear plant.

The addition to the first paragraph was made to emphasize the fact that the activities of the radiological environmental monitoring program are being reported rather than operation of the plant per se.

In the second paragraph the words "as appropriate" should apply to all of the items that might be specifically reported. Also, it is more appropriate to say "any observed impacts" as opposed to "the observed impacts" since impacts of plant operation are often miniscule, and thus, difficult to detect and confirm. Finally, the term "land use survey" is believed to be more appropriate than "land use census."

In the third paragraph, the inserted sentence is intended to prevent cluttering of the report with irrelevant data so that significant results will not be masked. The word "possible" was replaced with "practicable" on the basis that the submittal of missing data does not warrant the highest priority, especially when it may not be possible to substantiate that the priority was achieved.

In the last paragraph, the necessary maps will be provided in the reports and their number need not be specified in the Technical Specifications. Also, all area maps have been constructed with the midway point between the two reactors as the origin. The centers of the two reactors are a few hundred feet apart.

6.9.1.4:

The addition of the words "Historical annual average meteorological conditions or" is based on the fact that Section 3.3 of NUREG 0133 allows the use of historical annual average meteorological conditions for routine gaseous dose calculations.

The word "nearby" as it modifies "uranium fuel cycle sources" was deleted and the phrase "within 8 km" was inserted to provide clarification. This is consistent with the bases for Specification 3/4.11.4.

6.10.2, item "i":

This item was removed from Specification 6.10.3 and placed here because we feel that in a 5-year timeframe a sufficient amount of data will be available for us to identify any negative trends in secondary water quality.

6.10.3, item "i":

"QA Manual" was replaced with "FSAR" because Chapter 17 of the VEGP FSAR governs VEGP's Quality Assurance activities.

6.10.3 item "m":

See the justification for 6.10.2, item "i" for deletion of old item "m."

6.12.1:

This section was revised to clarify the requirements of the STS. These changes are consistent with the requirements of paragraph 12.3.1.2 of the VEGP FSAR and section 12.3.1 of the SER.

6.12.2:

The words "... equal to or less than 1000 mR/h..." were changed to "... greater than 100 mR/h but less than 1000 mR/h..." to be consistent with the definition of high radiation area provided in Part 20.202 of 10 CFR.

6.13.1 (STS):

This specification was deleted because the NRC has informed VEGP that we must have an approved PCP prior to receipt of a license. This direction was provided per a telephone conference call at 2:00 p.m. on 3/19/85. Therefore, 6.13.1 will be obsolete at the time of Technical Specification approval by the NRC, i.e., receipt of the VEGP Unit 1 License.

6.13.1.b:

The words "acceptance by the PRB" were deleted on the basis that the PRB makes recommendations to the GMVNO for approval of changes to the PCP.

6.14.1 (STS):

This specification was deleted because the NRC provided written direction in their "Summary of September 11, 1984, Technical Specification meeting for Vogtle, Units 1 and 2" that ". . . the offsite dose calculation manual must be approved by the staff prior to license issuance . . .". Therefore, 6.14.1 will be obsolete at the time of Technical Specification approval by NRC, i.e., receipt of the VEGP Unit 1 License.

6.14.1.b:

The words "acceptance by the PRB" were deleted on the basis that the PRB makes recommendations to the GMVNO for approval of changes to the ODCM.

6.15.1.a.8:

The words "by the PRB" were deleted on the basis that the PRB makes recommendations to the GMVNO for approval/disapproval.

6.16:

The justification for adding a specification to specify the date for which Technical Specifications changes are to be implemented is that these type of changes affect procedures and programs and time will be required to implement the changes (i.e., incorporate them into procedures and programs and train personnel). Also, it will take time to get all of the copies of the Technical Specifications posted to ensure that personnel are aware of the changes.

NUREG-xxx

TECHNICAL SPECIFICATIONS

VOGTLE ELECTRIC GENERATING PLANT

UNIT 1

DOCKET NOS. 50-424

APPENDIX "A" to

LICENSE NO. NPF-

issued by the
U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

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DEFINITIONS

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Technical Specification which prescribes remedial measures required under designated conditions.

ACTUATION LOGIC TEST

1.2 An ACTUATION LOGIC TEST shall be the application of various simulated input combinations in conjunction with each possible interlock logic state and verification of the required logic output. The ACTUATION LOGIC TEST shall include a continuity check, as a minimum, of output devices.

ANALOG CHANNEL OPERATIONAL TEST

1.3 An ANALOG CHANNEL OPERATIONAL TEST shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY of alarm, interlock and/or trip functions. The ANALOG CHANNEL OPERATIONAL TEST shall include adjustments, as necessary, of the alarm, interlock and/or Trip Setpoints such that the setpoints are within the required range and accuracy.

(normalized to their full power sum and expressed as their percentage)

AXIAL FLUX DIFFERENCE

1.4 AXIAL FLUX DIFFERENCE shall be the difference in ~~normalized~~ flux signals between the top and bottom halves of a two section excore neutron detector.

CHANNEL CALIBRATION

1.5 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel such that it responds within the required range and accuracy to known values of input. The CHANNEL CALIBRATION shall encompass the entire channel including the sensors and alarm, interlock and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.6 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions~~, except as provided in Table [3.6-1] of Specification [3.6.3]~~.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification ~~[~~3.6.1.3~~]~~,

d. The containment leakage rates are within the limits of Specification ~~[~~3.6.1.2~~]~~, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

~~CONTROLLED LEAKAGE~~

~~1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.~~

CORE ALTERATIONS

1.8 ~~1.9~~ CORE ALTERATIONS shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

DOSE EQUIVALENT I-131

1.9 ~~1.10~~ DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or~~ Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977~~]~~.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 ~~1.11~~ $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides ~~in the sample~~. other than radioiodines with half-lives greater than 14 minutes making up at least 95% of the total nonradioiodine activity in the sample.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.11 ~~1.12~~ The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.12 ~~1.13~~ The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

Insert 1.13

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except Reactor Coolant Pump Seal Leakoff ~~CONTROLLED LEAKAGE~~) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.15 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER(S) OF THE PUBLIC

1.16 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

INSERT 1.13

GASEOUS WASTE PROCESSING SYSTEM

1.13 A GASEOUS WASTE PROCESSING SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

OPERABLE - OPERABILITY

1.18 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter [14.0] of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and Federal and State regulations, burial ground requirements, and other requirements governing the disposal of radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3411 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage.

REPORTABLE EVENT

1.27 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

~~SHIELD BUILDING INTEGRITY~~

~~1.28 SHIELD BUILDING INTEGRITY shall exist when:~~

~~a. Each door in each access opening is closed except when the access opening is being used for normal transit entry and exit, then at least one door shall be closed,~~

~~b. The Shield Building Filtration System is OPERABLE, and~~

~~c. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.~~

SHUTDOWN MARGIN

1.28 ~~1.29~~ SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all ~~full-length~~ rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.29 ~~1.30~~ The SITE BOUNDARY shall be that ~~line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.~~ property line shown in Figure 5.1-1 of these Technical Specifications.

SLAVE RELAY TEST

1.30
~~1.31~~ A SLAVE RELAY TEST shall be the energization of each slave relay and verification of OPERABILITY of each relay. The SLAVE RELAY TEST shall include a continuity check, as a minimum, of associated testable actuation devices.

SOLIDIFICATION

1.31
~~1.32~~ SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

1.32
~~1.33~~ A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.33
~~1.34~~ A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.

THERMAL POWER

1.34
~~1.35~~ THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TRIP ACTUATING DEVICE OPERATIONAL TEST

1.35
~~1.36~~ A TRIP ACTUATING DEVICE OPERATIONAL TEST shall consist of operating the Trip Actuating Device and verifying OPERABILITY of alarm, interlock and/or trip functions. The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include adjustment, as necessary, of the Trip Actuating Device such that it actuates at the required Setpoint within the required accuracy.

UNIDENTIFIED LEAKAGE

1.36
~~1.37~~ UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE. ~~or CONTROLLED LEAKAGE~~.

UNRESTRICTED AREA

1.37
~~1.38~~ An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

1.38
~~1.39~~ A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.39
~~1.40~~ VENTING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

~~WASTE GAS HOLDUP SYSTEM~~

~~1.41 A WASTE GAS HOLDUP SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.~~

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 1.0

Definitions:

1.4:

This definition was revised so as to give definition to the term "normalized" and relieve a potentially ambiguous situation. Without qualification, the term normalized could imply that the flux signals be normalized to their sum, their full-power sum, or some other value.

1.7:

See the justification provided for Specification 4.6.1.1.a.

1.8:

The definition of CONTROLLED LEAKAGE was deleted as a result of changes made in specifications dealing with limiting flowrates through the RCP seal water injection line.

The purpose of the STS limit on controlled leakage is to ensure that the flow through the seal water injection line is less than that assumed in the accident analysis. This ensures that sufficient centrifugal charging pump injection flow is directed to the boron injection tank and, ultimately, to the RCS.

The generic STS surveillance (4.4.6.2.1.c) ensures the above by measuring CONTROLLED LEAKAGE at a specified RCS pressure with the modulating flow control valve (121) fully open. This method is sensitive to conditions in the chemical and volume control system, particularly the position of the charging flow valve (182). The proposed technical specification changes would provide a more quantitative measurement of the seal water line flow by limiting the flowrate across the RCP seal water injection throttle valves at a specified pressure drop across the valve. This will result in making the measurement independent of charging flow valve (182) position, modulating flow control valve (121) position, or CVCS system conditions.

The proposed surveillance requirement (4.5.2.g.3) would ensure proper RCP seal water injection throttle valve position and, hence, flow. The STS definition of CONTROLLED LEAKAGE was deleted since it is no longer necessary. The definitions of IDENTIFIED LEAKAGE and UNIDENTIFIED LEAKAGE were revised to account for the deletion of CONTROLLED LEAKAGE. Limits and surveillances on CONTROLLED LEAKAGE were deleted and a new surveillance requirement (4.5.2.g.3) implemented to control flow through the RCS seal water injection lines.

1.9:

This change was made to incorporate the correct reference for the dose conversion factors utilized in the VEGP safety analysis. See paragraph 1.9.109.2 and subsection 11.3.3 of the VEGP FSAR.

1.10:

The bases for Specification 3/4.4.8 state that the radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or greater than 14 minutes. The 10-minute half-life cutoff seems to be arbitrary especially in light of the fact that, depending on which reference you use, N-13 is reported to have a half-life of 9.97 minutes to 10.5 minutes. Establishing the cutoff at 14 minutes relieves this ambiguity while remaining consistent with the intent of the specification.

1.13:

The change in terminology (from waste gas holdup system to gaseous waste processing system) was made to reflect plant-specific nomenclature. See section 11.3 of the VEGP FSAR.

1.14:

The definition of IDENTIFIED LEAKAGE was revised to account for the deletion of CONTROLLED LEAKAGE. (See the justification for the deletion of the term CONTROLLED LEAKAGE.)

1.25 (STS):

The definition of SHIELD BUILDING INTEGRITY was deleted because it is not applicable to VEGP. The design of VEGP does not include a shield building.

1.28:

The words "full-length" were deleted on the basis that all RCCAs at VEGP are full-length. See paragraph 4.2.2.3 of the VEGP FSAR.

1.29:

This change was made to specifically identify only the land associated with the VEGP site and exclude other land owned by GPC adjacent to the site boundary.

1.36:

The term "CONTROLLED LEAKAGE" was deleted from the definition of UNIDENTIFIED LEAKAGE on the basis provided for the deletion of the definition of "CONTROLLED LEAKAGE."

1.41 (STS):

This definition now appears as 1.13 with the new title "Gaseous Waste Processing System." See the justification provided for definition 1.13.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

(PI-0455 A, B, + C, PI-0456 A, B, + C, PI-0457 A, B, + C, PI-0458 A, B, + C)

2.1 SAFETY LIMITS

REACTOR CORE (N-0041, N-0042, N-0043, N-0044)

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in Figure~~s~~ 2.1-1. ~~and 2.1-2 for n and n-1 loop operation, respectively~~.

Loop 1	TI-0412
Loop 2	TI-0422
Loop 3	TI-0432
Loop 4	TI-0442

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE (PT-0408, PT-0418, PT-0428, PT-0438)

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - FOUR LOOPS IN OPERATION

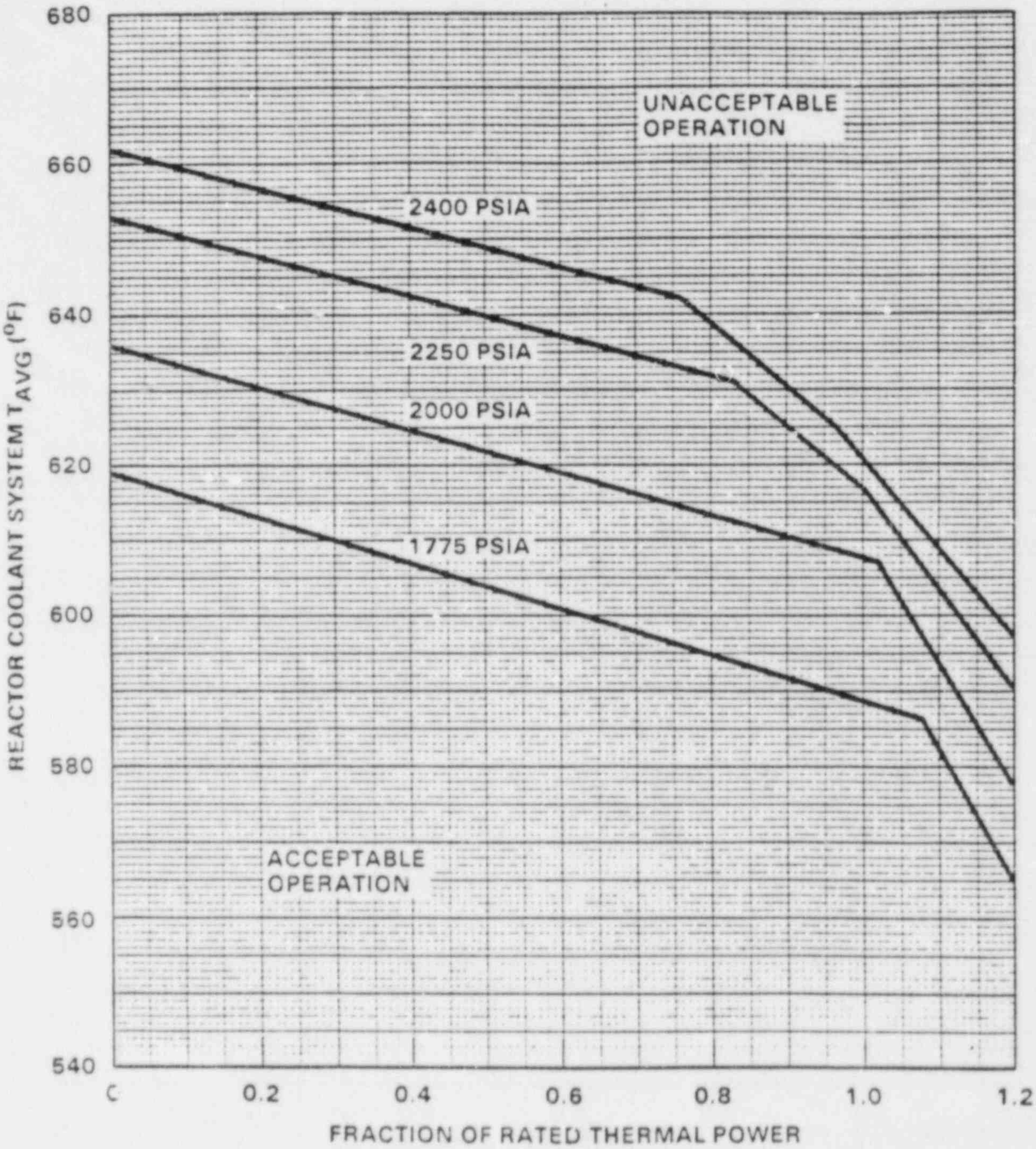

Figure 2.1-1. Reactor Core Safety Limits

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

a. With a Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 2.2-1, adjust the Setpoint consistent with the Trip Setpoint value.

b. With the Reactor Trip System Instrumentation or Interlock Setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 2.2-1 and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $$Z + R + S \leq TA$$

Where:

Z = The value from Column Z of Table 2.2-1 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 2.2-1 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 2.2-1 for the affected channel.

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux (N-0041, N-0042, N-0043, N-0044)					
a. High Setpoint	[7.5]	[4.56]	0	≤[109]% of RTP**	≤[~~111.2~~] 111.1% of RTP**
b. Low Setpoint	[8.3]	[4.56]	0	≤[25]% of RTP**	≤[~~27.2~~] 27.1% of RTP**
3. Power Range, Neutron Flux, High Positive Rate (N-0041, N-0042, N-0043, N-0044)	[~~2.0~~] 1.6	[0.5]	0	≤[5]% of RTP** with a time constant ≥[2] seconds	≤[~~6.8~~] 6.3% of RTP** with a time constant ≥[2] seconds
4. Power Range, Neutron Flux, High Negative Rate (N-0041, N-0042, N-0043, N-0044)	[~~2.0~~] 1.6	[0.5]	0	≤[5]% of RTP** with a time constant ≥[2] seconds	≤[~~6.8~~] 6.3% of RTP** with a time constant ≥[2] seconds
5. Intermediate Range, Neutron Flux (N-0035, N-0036)	[17.0]	[8.41]	0	≤[25]% of RTP**	≤[~~31~~] 30.9% of RTP**
6. Source Range, Neutron Flux (N-0031, N-0032)	[17.0]	[10.01]	0	(Later) ≤[~~10^5~~] cps	(Later) ≤[~~1.4 x 10^5~~] cps
7. Overtemperature ΔT (TE-0411 A+B, TE-0421 A+B, TE-0431 A+B, TE-0441 A+B)	[~~6.7~~] 6.5	[~~2.79~~] 3.4	[0.8]	See Note 1	See Note 2
8. Overpower ΔT (TE-0411 A+B, TE-0421 A+B, TE-0431 A+B, TE-0441 A+B)	[~~4.3~~] 5.0	1.43 [~~1.3~~]	[0.2]	See Note 3	See Note 4
9. Pressurizer Pressure-Low (PI-0455 A,B,+C, PI-0456 A,B,+C, PI-0457 A,B,+C, PI-0458 A,B,+C)	[~~6.8~~] 3.1	[0.71]	[1.5]	≥[~~1900~~] 1960 psig	≥[~~1886~~] 1948 psig
10. Pressurizer Pressure-High (PI-0455 A,B,+C, PI-0456 A,B,+C, PI-0457 A,B,+C, PI-0458 A,B,+C)	[3.1]	[0.71]	[1.5]	≤[2385] psig	≤[239~~8~~7] psig
11. Pressurizer Water Level-High (LT-0459, LT-0460, LT-0461)	[5.0]	[2.18]	[1.5]	≤[92]% of instrument span	≤[93.8]% of instrument span
12. Reactor Coolant Flow-Low	[2.5]	[~~1.0~~] 0.50	[1.5]	≥[90]% of loop design flow*	≥[~~89.2~~] 88.7% of loop design flow*

*Loop design flow = [95,700] gpm
**RTP = RATED THERMAL POWER

Loop 1	Loop 2	Loop 3	Loop 4
FT-0414	FT-0424	FT-0434	FT-0444
FT-0415	FT-0425	FT-0435	FT-0445
FT-0416	FT-0426	FT-0436	FT-0446

Loop-1 Loop-2 Loop-3 Loop-4

LT-0517	LT-0527	LT-0537	LT-0547
LT-0518	LT-0528	LT-0538	LT-0548
LT-0519	LT-0529	LT-0539	LT-0549
LT-0551	LT-0552	LT-0553	LT-0554

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	~~[30.0]~~ 17.0	~~[27.18]~~ 12.18	[1.5]	≥~~[32.3]~~ 17.0 % of narrow range instrument span	≥~~[30.4]~~ 15.3 % of narrow range instrument span
~~14. Steam/Feedwater Flow Mismatch Coincident With~~	~~[16.0]~~	~~[13.24]~~	~~[1.5]~~	~~<[40]% of full steam flow at RTP**~~	~~<[42.5]% of full steam flow at RTP**~~
~~Steam Generator Water Level-Low-Low~~	~~[12.0]~~	~~[9.18]~~	~~[1.5]~~	~~<[32.3]% of narrow range instrument span~~	~~<[30.4]% of narrow range instrument span~~
14 ~~15~~. Undervoltage - Reactor Coolant Pumps	(Later) ~~[2.0]~~	(Later) ~~[1.28]~~	(Later)	≥~~[4830]~~ 9660 volts AC (70% bus voltage)	≥~~[4760]~~ 9522 volts AC (69% bus voltage)
15 ~~16~~. Underfrequency - Reactor Coolant Pumps	(Later) ~~[7.5]~~	(Later) ~~0~~	(Later) ~~[0.1]~~	≥~~[57.5]~~ Hz 57.2	≥~~[57.1]~~ Hz 57.1
16 ~~17~~. Turbine Trip					
a. Low Fluid Oil Pressure (PT-6161, PT-6162, PT-6163)	N.A.	N.A.	N.A.	≥~~[800]~~ 600 psig	≥~~[750]~~ 500 psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥~~[1]~~ 97.6 % open	≥~~[1]~~ 97.6 % open
17 ~~18~~. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

~~**RTP = RATED THERMAL POWER~~

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18 ~~19~~. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (N-0035, N-0036)	N.A.	N.A.	N.A.	≥[1 x 10^{-10}] amps	≥[6 x 10^{-11}] amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input (N-0041, N-0042, N-0043, N-0044)	N.A.	N.A.	N.A.	≤[10]% of RTP**	≤[12.~~2~~ 1]% of RTP**
2) P-13 input (PT-0505, PT-0506)	N.A.	N.A.	N.A.	≤[10]% RTP** Turbine Impulse Pressure Equivalent	≤[12.~~2~~ 1]% RTP** Turbine Impulse Pressure Equivalent
c. Power Range Neutron Flux, P-8 (N-0041, N-0042, N-0043, N-0044)	N.A.	N.A.	N.A.	~~≤[48]% of RTP**~~ (Later)	~~≤[50.2]% of RTP**~~ (Later)
d. Power Range Neutron Flux, P-9 (N-0041, N-0042, N-0043, N-0044)	N.A.	N.A.	N.A.	≤[50]% of RTP**	≤[52.~~2~~ 1]% of RTP**
e. Power Range Neutron Flux, P-10 (N-0041, N-0042, N-0043, N-0044)	N.A.	N.A.	N.A	≥[10]% of RTP**	≥[7.~~8~~ 9]% of RTP**
f. Turbine Impulse Chamber Pressure, P-13 (PT-0505, PT-0506)	N.A.	N.A.	N.A.	≤[10]% RTP** Turbine Impulse Pressure Equivalent	≤[12.~~2~~ 1]% RTP** Turbine Impulse Pressure Equivalent
19 ~~20~~. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20 ~~21~~. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \cancel{=} \geq$ [8] s, $\tau_2 \cancel{=} \leq$ [3] s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, $\tau_3 \cancel{=} \leq$ 0 ~~[2]~~ s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10 ~~[1.09]~~;
K_2	~~=~~	~~[0.0138]/°F~~; $\geq$ 0.012/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_4 \cancel{=} \geq$ 28 ~~[33]~~ s, $\tau_5 \cancel{=} \leq$ [4] s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \cancel{=} \leq$ 0 ~~[2]~~ s;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	[588.5]°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	~~=~~	~~[0.000671]/psig~~; $\leq$ 0.00056/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between ~~-[35]%~~ -33.5% and + ~~[7]%~~ 6.5%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~-[35]%~~ -33.5%, the ΔT Trip Setpoint shall be automatically reduced by ~~[1.26]%~~ 1.27% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds ~~+[7]%~~ +6.5%, the ΔT Trip Setpoint shall be automatically reduced by ~~[1.05]%~~ 0.83% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~[3.8]%~~ 2.6%.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1+\tau_7 S}\right)\left(\frac{1}{1+\tau_6 S}\right) T - K_6 [T \left(\frac{1}{1+\tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where:

ΔT = ~~As defined in Note 1,~~ Measured ΔT by RTD Manifold Instrumentation,

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = ~~As defined in Note 1,~~ Lead-Lag compensator on measured ΔT,

τ_1, τ_2 = ~~As defined in Note 1,~~ Time constants utilized in lead-lag compensator for ΔT, $\tau_1 \ge 8$ s, $\tau_2 \le 3$ s,

$\frac{1}{1+\tau_3 S}$ = ~~As defined in Note 1,~~ Lag compensator on measured ΔT,

τ_3 = ~~As defined in Note 1,~~ Time constants utilized in the lag compensator for ΔT, $\tau_3 \le 0$ s.

ΔT_o = ~~As defined in Note 1,~~ Indicated ΔT at RATED THERMAL POWER,

K_4 ≤ ~~=~~ ~~[1.09]~~, 1.089

K_5 ≥ ~~=~~ [0.02]/°F for increasing average temperature and ≥ 0 for decreasing average temperature,

$\frac{\tau_7 S}{1+\tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 ≥ ~~=~~ [10] s,

$\frac{1}{1+\tau_6 S}$ = ~~As defined in Note 1,~~ Lag compensator on measured T_{avg},

τ_6 = ~~As defined in Note 1,~~ Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 \le 0$ s,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

$\geq$ 0.0013/°F

K_6 = ~~[0.00128]/°F~~ for T > T" and K_6 $\geq$ ~~=~~ 0 for T $\leq$ T",

T = ~~As defined in Note 1,~~ Average Temperature, °F,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ [588.5]°F),

S = ~~As defined in Note 1, and~~ Laplace transform operator s^{-1}, and

$f_2(\Delta I)$ = 0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~[3.0]%~~.
3.6% of ΔT span.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 2.0

2.1:

Instrument tag numbers were added, as appropriate, to assist the operator in ensuring compliance with these Technical Specifications.

2.1.1:

The deletions reflect the intent of the plant not to operate in N-1 loop operation.

Table 2.2-1:

The following functional unit was deleted because it is not applicable to the VEGP reactor trip system design:

14. Steam generator water level low coincident with steam feedwater flow mismatch.

Table 2.2-1 (Notations):

Values provided represent VEGP plant design (hardware, software, and Statistical Setpoint Study Analysis results).

Note 3 - The words "As defined in Note 1" were deleted and the respective definitions were added to provide clarification.

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The BASES contained in succeeding pages summarize the reasons for the Specifications in Section 2.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB through the W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The minimum value of the DNBR during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figures [2.1-1] ~~and [2.1-2]~~ the limit value show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than 1.30, or ~~the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.~~ Insert for Page B2-1 here

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.55\ [1 + \overset{0.3}{\cancel{0.2}}\ (1-P)]$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

the combination of THERMAL POWER, Reactor Coolant System pressure, and average temperature are outside of the range for which the W-3 correlation applies.

The curves of Figure 2.1-1 are a result of the intersection of three lines which have the same Reactor Coolant pressure but have variable THERMAL POWER and average temperature. The most limiting line segments developed provide the curves of Figure 2.1-1. From right to left along the curves of Figure 2.1-1 the line segments provide the following THERMAL POWER, Reactor Coolant System pressure, and average temperature loci of points of which:

A. The W-3 correlation applies

- the minimum DNBR is no less than the limit value (far right line segment);

B. The W-3 correlation does not apply

- the exit quality of the hottest channel enthalpy is not greater than 15% (middle line segment on Reactor Coolant System pressure curves, 2400 psia and 2250 psia; and

- the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid (far left line segment).

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (~~3110~~ 3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than 1.30.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Insert to page B2-5 here

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

~~Optional for Plants Permitted n-1 Loop Operation~~

~~Operation with a reactor coolant loop out of service below the (n) loop P-8 Setpoint does not require Reactor Trip System Setpoint modification because the P-8 Setpoint and associated trip will prevent DNB during (n-1)~~ loop operation exclusive of the ~~Overtemperature ΔT Setpoint. (n-1) loop operation above the (n) loop P-8 Setpoint is permissible after resetting the K1 input to the Overtemperature ΔT channels and raising the P-8 Setpoint to its (n-1) loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level~~.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet *cracking or* melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of

Insert to page B 2-5

No credit was taken for operation of the trips associated with either the Intermediate or Source Range channel in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overpower ΔT (Continued)

change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately ~~49%~~ of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

~~Optional for Plants Permitted n-1 Loop Operation~~

~~The P-8 Setpoint trip will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients when [n-1] loops are in operation and the Overtemperature ΔT Trip Setpoint is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT Trip Setpoint adjusted to the value specified for [n-1] loop operation, the P-8 trip at [76%] RATED THERMAL POWER will prevent the minimum value of the DNBR from going below 1.30 during normal operational transients and anticipated transients with [n-1] loops in operation.~~

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

~~Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level~~

~~The Steam/Feedwater Flow Mismatch in coincidence with a Steam Generator Low Water Level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Trip System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to [1.42 x 10^6] lbs/hour. The Steam Generator Low Water level portion of the trip is activated when the water level drops below [25]%, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a Reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.~~

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the ~~simultaneous trip of two or more~~ loss of power to the reactor coolant pump bus circuit breakers shall not exceed [1.2] seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed [0.3] second.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses (Continued)

On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System Instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), ~~provides a backup block for Source Range Neutron Flux doubling,~~ and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump bus undervoltage and underfrequency, ~~Turbine trip,~~ pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

Reactor Trip System Interlocks (Continued)

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops, ~~and one or more reactor coolant pump breakers open.~~ On decreasing power, the P-8 automatically blocks the above ~~listed~~ trips.

P-9 On increasing power, P-9 automatically enables Reactor trip on Turbine trip. On decreasing power, P-9 automatically blocks Reactor trip on Turbine trip.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 2.0 BASES

General:

These bases have been revised to be consistent with the Technical Specifications as marked in this draft. In addition, certain revisions have been made for the purposes of amplification and clarification.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding specifications is required during the OPERATIONAL MODES or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications.

This specification is not applicable in MODE 5 or 6.

3.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual specifications.

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be met during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

Replace with insert 4.0.2.

4.0.2 ~~Each Surveillance Requirement shall be performed within the specified time interval with~~:

a. ~~A maximum allowable extension not to exceed 25% of the surveillance interval, but~~

b. ~~The combined time interval for any three consecutive surveillance intervals shall not exceed 3.25 times the specified surveillance interval~~.

4.0.3 Failure to perform a Surveillance Requirement* within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation has been performed within the stated surveillance interval or as otherwise specified.

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 inspection of ASME Code Class 1, 2, and 3 components and 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 Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i);

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*Upon determination that the Surveillance Requirement had been inadvertently omitted, power operation may continue provided that the Surveillance Requirement shall be successfully performed within the Limiting Condition for Operation (LCO) period, which would begin upon discovery.

Insert 4.0.2

4.0.2 Each Surveillance Requirement shall be performed within the specified time interval. Extensions shall not exceed the following:

Specified Interval	Maximum Time Between Any Two Surveillances	Maximum Time For Any Four Consecutive Surveillances
30 min.	38 min.	98 min.
60 min.	75 min.	195 min.
2 hrs.	2.5 hrs.	6.5 hrs.
4 hrs.	5 hrs.	13 hrs.
12 hrs.	16 hrs.	39 hrs.
24 hrs.	30 hrs.	78 hrs.
72 hrs.	90 hrs.	234 hrs.
7 days	9 days	23 days
14 days	18 days	46 days
31 days	39 days	101 days
92 days	115 days	299 days
184 days	230 days	598 days
366 days	458 days	1190 days
18 mos.	23 mos.	59 mos.
3 yrs.	3.75 yrs.	9.75 yrs.
40 mos.	50 mos.	130 mos.
5 yrs.	6.25 yrs.	16.25 yrs.
10 yrs.	12.5 yrs.	32.5 yrs.

SURVEILLANCE REQUIREMENTS (Continued)

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:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities;

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements; and

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.0

4.0.2:

VEGP proposes to replace the formula presented in Specification 4.0.2 with a table which presents each surveillance interval specified in the Technical Specifications; the maximum time allowed between any two surveillances; and the maximum time allowed for any four consecutive surveillances. Since the formula presented in Specification 4.0.2 has been applied to arrive at the tabular form, the two specifications are identical in intent. The proposed revision has the advantage of eliminating ambiguity.

4.0.3, Footnote *:

The addition of this footnote allows the time interval for the LCO to begin upon discovery that a surveillance requirement had been inadvertently omitted rather than when the surveillance should have been performed. If a piece of equipment, a system, or an instrument, as the case may be, has been demonstrated operable by the performance of the previous surveillances, there is no reason (other than a legalistic argument) to assume the inoperability of the equipment on the basis that the next required surveillance was inadvertently omitted. By allowing the time period for the LCO to begin upon discovery, the licensee has the opportunity to demonstrate the operability of the equipment. Otherwise, the operator could be placed in the position of arbitrarily declaring a piece of equipment inoperable and possibly shutting the plant down when in fact the equipment was operable and capable of performing its function. This interpretation has been approved and incorporated by the NRC in the Farley Plant Technical Specifications.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} GREATER THAN 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to: ~~[1.6%] Δk/k for [n] loop operation~~.
a. 1.3 % delta - K/K;
b. (Later) delta - K/K.

APPLICABILITY: ~~MODES 1, 2*, 3, and 4~~.
Specification 3.1.1.1(a) - MODES 1 and 2*
Specification 3.1.1.1(b) - MODES 3 and 4

ACTION:

With the SHUTDOWN MARGIN less than ~~[1.6%] Δk/k~~ the above Specification 3.1.1.1(a) and (b), immediately initiate and continue boration at greater than or equal to 30 gpm (FT-0110, FT-0111, FT-0183) of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal ~~to [1.6%] Δk/k~~: to the above Specification 3.1.1.1(a) and (b):

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above Specification 3.1.1.1(a) and (b) required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. ~~When in MODE 2~~ With K_{eff} less than 1, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6;

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of Specification 4.1.1.1.1e. below, with the control banks at the maximum insertion limit of Specification 3.1.3.6; and

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*See Special Test Exceptions Specification 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3 or 4, at least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification [4.1.1.1.1e.], above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to ~~1%~~ (Later) $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than ~~1%~~ (Later) $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 30 gpm (FT-0110, FT-0111, FT-0183) of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1%~~ (Later) $\Delta k/k$:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than [0] Δk/k/°F for the all rods withdrawn, beginning of cycle life (BOL), hot zero THERMAL POWER condition; and

b. Less negative than -[3.9] x 10^{-4} Δk/k/°F for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only**.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 Δk/k/°F within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

   2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

   3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

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*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to -[3.0] x 10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -[3.0] x 10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to ~~[541]~~ 551°F.

Loop 1 TI-0412
Loop 2 TI-0422
Loop 3 TI-0432
Loop 4 TI-0442

APPLICABILITY: MODES 1 and 2* **.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 551 → ~~[541]~~°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to ~~[541]~~ 551°F:

a. Within 15 minutes prior to achieving reactor criticality, and

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than ~~[551]~~ 561°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

(TI-0412, TI-0422, TI-0432, TI-0442)

*With K_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE. ~~and capable of being powered from an OPERABLE emergency power source~~:

a. A flow path from the boric acid storage tank via ~~either~~ a boric acid transfer pump ~~or a gravity feed connection~~ and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification [3.1.2.5a.] is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification [3.1.2.5b.] is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE ~~or capable of being powered from an OPERABLE emergency power source~~, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. ~~At least once per 7 days by verifying that the temperature [of the heat traced portion] of the flow path is greater than or equal to [145]°F when a flow path from the boric acid tanks is used, and~~

At least once per 7 days when the boric acid storage tank is the required water source, verify that the applicable portions of the auxiliary building (TISL 12410 or TISL 12411, TISL 12412 or TISL 12413, TISL 12414 or TISL 12415, TISL 12416 or TISL 12417, TISL 20900 or TISL 20901, TISL 20902 or TISL 20903, TISL 20904 or TISL 20905) and the portions of the flow path for which ambient temperature indication are not provided are ≥ 65 °F.

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two* of the following three boron injection flow paths shall be OPERABLE:

a. ~~The~~ A flow path from the boric acid storage tank~~s~~ via a boric acid transfer pump and a charging pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging pumps to the RCS.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1%~~ $\Delta k/k$ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

(Later)

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

Replace with insert for page 3/4 1-8

a. ~~At least once per 7 days by verifying that the temperature [of the heat traced portion] of the flow path from the boric acid tanks is greater than or equal to [145]°F when it is a required water source~~;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position; and

~~c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a __ test signal; and~~

c. ~~d.~~ At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

~~*Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F.~~

Insert for Page 3/4 1-8

At least once per 7 days when the boric acid storage tank is a required water source, verify that the applicable portions of the auxiliary building (TISL 12410 or TISL 12411, TISL 12412 or TISL 12413, TISL 12414 or 12415, TISL 12416 or TISL 12417, TISL 20900 or TISL 20901, TISL 20902 or TISL 20903, TISL 20904 or TISL 20905) and the portions of the flow path for which ambient temperature indication are not provided are $\geq 65^{\circ}F$.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least ~~One~~ one charging pump in the boron injection flow path required by Specification ~~[~~3.1.2.1~~]~~ shall be OPERABLE. ~~and capable of being powered from an OPERABLE emergency power source~~.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump OPERABLE, ~~or capable of being powered from an OPERABLE emergency power source,~~ suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

~~4.1.2.3.1 The above required charging pump shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across the pump of greater than or equal to ____ psid is developed when tested pursuant to Specification 4.0.5~~.

~~4.1.2.3.2 All charging pumps, excluding the above required OPERABLE pump~~, ~~shall be demonstrated inoperable at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position~~.

4.1.2.3 At least once per 18 months the above required charging pump shall be demonstrated OPERABLE by verifying that the flow path required by Specification 3.1.2.1 is capable of delivering at least 30 GPM to the RCS.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two* charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least ~~1%~~ (Later) $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

~~4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying, on recirculation flow, that a differential pressure across each pump of greater than or equal to ____ psid is developed when tested pursuant to Specification 4.0.5~~.

~~4.1.2.4.2 All charging pumps, except the above allowed OPERABLE pump, shall be demonstrated inoperable at least once per 31 days whenever the temperature of one or more of the Reactor Coolant System (RCS) cold legs is less than or equal to [275]°F by verifying that the motor circuit breakers are secured in the open position~~.

---

*~~A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F~~.

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.1.2.2.c.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage ~~System~~ Tank with:

1) A minimum contained borated water volume of (Later) gallons, (LI-102A, LI-104A),

2) A boron concentration between [7000] ppm and ~~[8000]~~ 7700 ppm, and

3) A minimum solution temperature of [65]°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of (Later) gallons, (LI-990, LI-991, LI-992, LI-993)

2) A boron concentration between [2000] ppm and [2200] ppm, and

3) A minimum solution temperature of ~~[35]~~ 50°F. (TIS-10980)

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the contained borated water volume, and

3) ~~Verifying the boric acid storage tank solution temperature when it is the source of borated water~~.

b. At least once per 24 hours by verifying the RWST temperature (TIS-10980) when it is the source of borated water and the ~~[outside]~~ air temperature is less than ~~[35]~~ 50°F.

When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20900) is < 65°F, verify the boric acid storage tank solution temperature is ≥ 65°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. A Boric Acid Storage ~~System~~ Tank with:

1) A minimum contained borated water volume of (Later) gallons, (LI-102A, LI-104A),

2) A boron concentration between [7000] ppm and [~~8000~~] 7700 ppm, and

3) A minimum solution temperature of [65]°F.

b. The refueling water storage tank (RWST) with:

1) A minimum contained borated water volume of (Later) gallons, (LI-990, LI-991, LI-992, or LI-993)

2) A boron concentration between [2000] ppm and [~~2200~~] 2100 ppm, and

3) A minimum solution temperature of [~~35~~] 50 °F, and

4) A maximum solution temperature of [~~100~~] 120 °F, (TIS-10980)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

(Later)

a. With the Boric Acid Storage ~~System~~ Tank inoperable and being used as one of the above required borated water sources, restore the ~~system~~ Tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least ~~1%~~ Δk/k at 200°F; restore the Boric Acid Storage ~~System~~ to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST Tank inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the contained borated water volume of the water source, and

3) ~~Verifying the Boric Acid Storage System solution temperature when it is the source of borated water~~.

(TIS-10980)

b. At least once per 24 hours by verifying the RWST temperature when the [outside] air temperature is ~~either~~ less than [~~35~~]°F ~~or greater than [100]°F~~.
50

When the boric acid storage tank is the source of borated water and the ambient temperature of the boric acid storage tank room (TISL-20902, TISL-20900) is ≤ 65°F, verify the boric acid storage tank solution temperature is ≥ 65°F.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full-length shutdown and control rods shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full-length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full-length rod inoperable or misaligned from the group step counter demand position by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full-length rod trippable but inoperable due to causes other than addressed by ACTION a., above, or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within 1 hour:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figures [3.1-1] and [3.1-2]. The THERMAL POWER level shall be restricted pursuant to Specification [3.1.3.6] during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions;

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours;

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All shutdown and control rods shall be OPERABLE and positioned within $\pm$ 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* AND 2*.

ACTION:

a. With one or more rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With one rod trippable but inoperable due to causes other than addressed by ACTION a above, or misaligned from its group step counter demand height by more than $\pm$12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the remainder of the rods in the group with the inoperable rod are aligned to within $\pm$12 steps of the inoperable rod while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

   3. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) A reevaluation of each accident analysis of Table 3.1-1 is performed within 5 days; this reevaluation shall confirm that the previously analyzed results of these accidents remain valid for the duration of operation under these conditions.

      b) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

* See Special Test Exceptions Specification 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c) A power distribution map is obtained from the movable incore detectors and $F_Q(Z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours; and

d) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full-length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full-length rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

c) The THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within the next hour and within the following 4 hours the High Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER.

c. With more than one rod trippable but inoperable due to causes other than addressed by ACTION a above, POWER OPERATION may continue provided that:

1. Within one hour, the remainder of the rods in the bank(s) with the inoperable rods are aligned to within ± 12 steps of the inoperable rods while maintaining the rod sequence and insertion limits of Figure 3.1-1. The THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, and

2. The inoperable rods are restored to OPERABLE status within 72 hours.

d. With more than one rod misaligned from it's group step counter demand height by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each rod not fully inserted in the core shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE FULL-LENGTH ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates the Emergency Core Cooling System

Single Rod Cluster Control Assembly Withdrawal at Full Power

Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accident)

Major Secondary Coolant System Pipe Rupture

Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

TABLE 3.1-1

ACCIDENT ANALYSES REQUIRING REEVALUATION
IN THE EVENT OF AN INOPERABLE CONTROL OR
SHUTDOWN ROD

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

Decrease in Reactor Coolant Inventory

Inadvertent Opening of a Pressurizer Safety or Relief Valve

Break in Instrument Line or Other Lines from Reactor Coolant Pressure Boundary That Penetrate Containment

Loss-of-Coolant-Accidents

Increase in Heat Removal by the Secondary System (Steam System Piping Rupture)

Spectrum of Rod Cluster Control Assembly Ejection Accidents.

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 The Digital Rod Position Indication System and the Demand Position Indication System shall be OPERABLE and capable of determining the control *and shutdown* rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one digital rod position indicator per bank inoperable either:

   1. Determine the position of the nonindicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the nonindicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all digital rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

c. The provisions of Specification 3.0.4 are not applicable when verifying system operability following repair.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each digital rod position indicator shall be determined to be OPERABLE by verifying that the Demand Position Indication System and the Digital Rod Position Indication System agree within 12 steps at least once per 12 hours except during time intervals when the rod position deviation monitor is inoperable, then compare the Demand Position Indication System and the Digital Rod Position Indication System at least once per 4 hours.

POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 One digital rod position indicator (excluding demand position indication) shall be OPERABLE and capable of determining the control rod position within ± 12 steps for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3* **, 4* **, and 5* **.

ACTION:

With less than the above required position indicator(s) OPERABLE, immediately open the Reactor Trip System breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required digital rod position indicator(s) shall be determined to be OPERABLE by verifying that the digital rod position indicators agree with the demand position indicators within 12 steps when exercised over the full-range of rod travel, at least once per 18 months.

---

*With the Reactor Trip System breakers in the closed position.

**See Special Test Exceptions Specification 3.10.5.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual ~~full-length~~ (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to [2.2] seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

(TI-0412, TI-0422, TI-0432, TI-0442)

a. T_{avg} greater than or equal to ~~[541]~~ 551°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any ~~full-length~~ rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

~~b. With the rod drop times within limits but determined with n-1 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to either:~~

~~1. Less than or equal to [66]% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or~~

~~2. Less than or equal to [76]% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.~~

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of ~~full-length~~ rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically 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. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification [4.1.3.1.2], within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification [3.1.3.1].

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as shown in Figures [3.1-1]. ~~and [3.1-2]~~.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the above insertion limits, except for surveillance testing pursuant to Specification [4.1.3.1.2]:

a. Restore the control banks to within the limits within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the above figure~~s~~, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

FIGURE 3.1-1

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
FOUR LOOP OPERATION

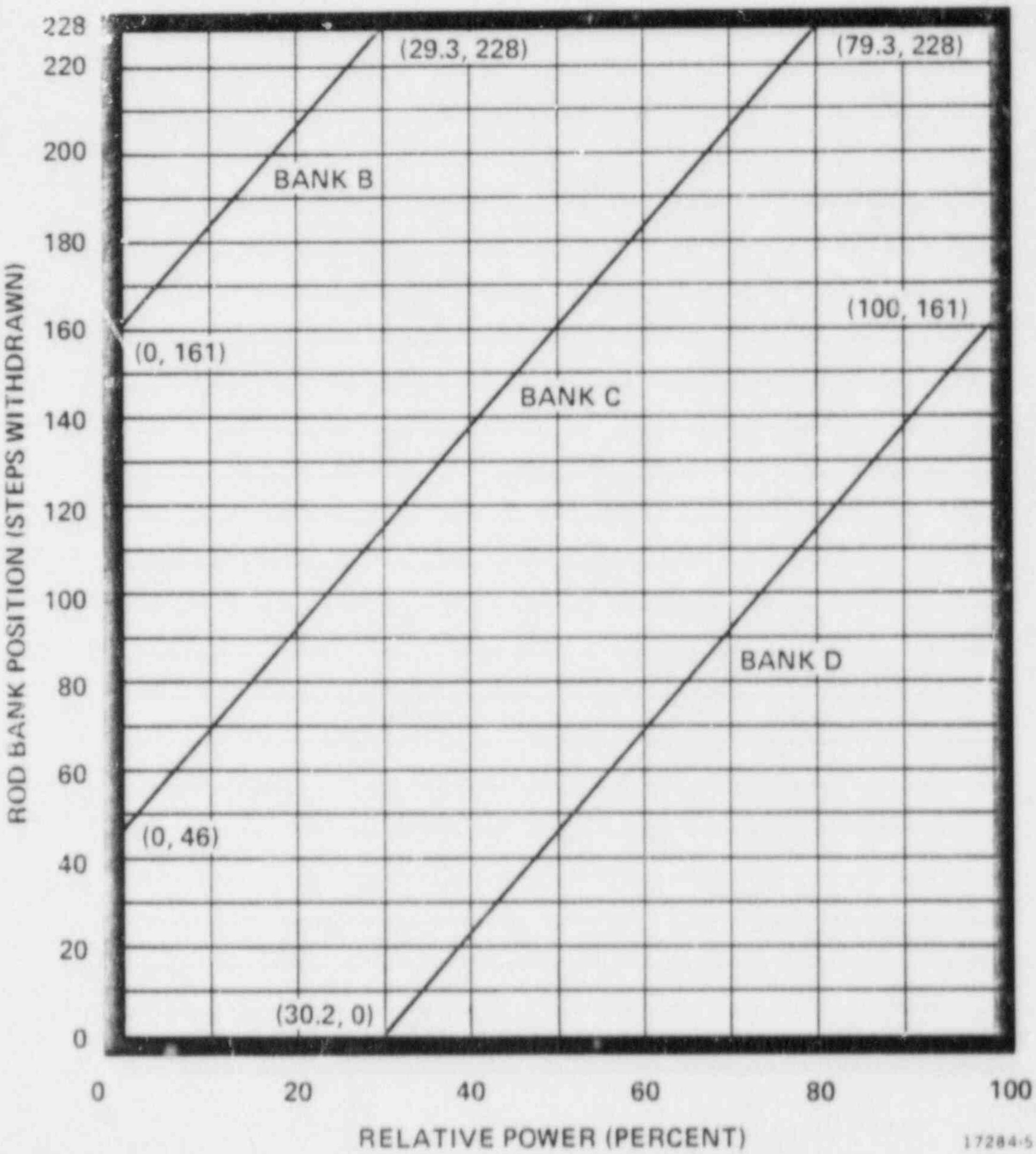


Figure 3.1-1. Rod Bank Insertion Limits Versus Thermal Power

FIGURE 3.1-2

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
THREE LOOP OPERATION

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.1

3/4.1:

Instrument tag numbers were added as appropriate to assist the operator in ensuring compliance with these Technical Specifications.

3/4.1.1.1:

The specification was revised to reflect the FSAR accident analysis results requiring an SDM of 1.3% delta-K/K for MODES 1 and 2 and (Later) delta-K/K for Modes 3 and 4.

3.1.1.1:

Reference to "(N) loop operation" was deleted. This reflects the intent of the plant not to operate in N-1 loop operation. Removal of this statement also eliminates confusion when operating below P-8 while under Action Statement 3.4.1.1.

4.1.1.1.1.c:

The words "when in MODE 2" were deleted on the following basis.

MODE 2 is normally achieved during critical approach, i.e., while the control banks are being withdrawn. To verify that predicted critical control rod position is within limits (when in Mode 2 with Keff < 1.0) would be disruptive to the critical approach.

As proposed, the 4-hour surveillance requirement provides sufficient assurance that adequate shutdown margin will be maintained until criticality is achieved, at which time the "estimated critical condition calculation" and rod insertion limit alarms will immediately ensure that critical control rod position is within limits.

3/4.1.12:

The specification was revised to reflect the FSAR accident analysis (boron dilution accident) requiring an SDM of (Later) delta-K/K for MODE 5.

3.1.2.1:

The VEGP design includes one boric acid storage tank per unit with two boric acid transfer pumps. The VEGP design does not take credit for gravity feed. See paragraph 9.3.4.1.2.3 of the VEGP FSAR.

Reference to an operable emergency power source was removed on the strength of an analysis performed to demonstrate compliance with section 15.4.6 of the Standard Review Plan (SRP). Section 15.4.6 of the SRP addresses moderator dilution transients. Moderator dilution events in Mode 6 are precluded at VEGP on the basis that the source of pure water is "locked out." Likewise, administrative controls will be used to preclude the initiators identified in the analysis for Mode 5. The result is that the probability of a moderator dilution event is greatly reduced for these modes. In addition, the inherent stability of the Southern electric system grid is such that the probability of a loss-of-offsite power in conjunction with a moderator dilution event in Modes 5 or 6 should be sufficiently small as to be negligible.

4.1.2.1.a:

The flowpath from the boric acid storage tank to the centrifugal charging pump suction header is routed through portions of the auxiliary building which are maintained at a temperature ≥ 65°F. Therefore, as long as these portions of the auxiliary building are maintained at a temperature ≥ 65°F, the temperature of the flowpath will also be ≥ 65° and the flowpath temperature should not require monitoring. The ambient temperature of these portions of the auxiliary building is monitored by redundant temperature instrumentation and annunciated in the control room. The temperature of those portions of the flow path which are not routed through a controlled environment will be monitored once per 7 days when the boric acid storage tank is the required water source. See paragraph 9.3.4.1.2.3 of the VEGP FSAR.

3.1.2.2:

The VEGP design includes one boric acid storage tank per unit with two boric acid transfer pumps. There are also two flowpaths from the boric acid storage tank, either of which will meet the intent of this specification. See paragraph 9.3.4.1.2.3 of the VEGP FSAR.

The action statement shutdown margin requirement was revised to reflect the FSAR accident analysis results (boron dilution accident) which require a shutdown margin of (Later) delta-K/K at 200°F.

Footnote * was deleted on the basis that both centrifugal charging pumps and the positive displacement pump are allowed to be operable below the cold overpressure protection system (COPS) arming temperature of 350°F. Therefore, footnote * is not needed. See paragraph 5.2.2.10 of the VEGP FSAR.

4.1.2.2.a:

See the justification for 4.1.2.1.a.

4.1.2.2.c (STS):

The automatic valves in the flowpaths from the refueling water storage tank to the reactor coolant system are demonstrated OPERABLE as per Specification 4.5.2.e.1.

3.1.2.3:

The words "at least" were added since the cold overpressurization analysis assumes the operability of two centrifugal charging pumps and the positive displacement pump. See the justification provided for 3.1.2.1 for deleting reference to an operable emergency power source.

4.1.2.3.1:

The intent of this specification is to ensure the capability to offset a positive reactivity addition accident. By verifying that the required charging pump(s) will deliver at least 30 gpm to the RCS, the necessary negative reactivity insertion rate should be assured. In this case, pump operability need not be based on the development of a specified differential pressure.

4.1.2.3.2:

This surveillance was deleted since both centrifugal charging pumps and the positive displacement pump are allowed to be operable below the COPS arming temperature of 350°F. See paragraph 5.2.2.10 and 5.2.2.11 of the VEGP FSAR.

3.1.2.4:

Footnote * was deleted for the reason discussed under 4.1.2.3.2, above.

The action statement shutdown margin requirement was revised to reflect the FSAR accident analysis results (boron dilution accident) which requires a shutdown margin of (Later) delta-K/K at 200°F.

4.1.2.4.1:

See the justification provided for 4.1.2.3.1.

4.1.2.4.2:

See the justification provided for 4.1.2.3.2.

3.1.2.5.a:

For the purposes of this specification, the boric acid storage "system" consists of the boric acid storage tank, only. See paragraph 9.3.4.1.2.3 of the VEGP FSAR.

4.1.2.5.a.3:

The temperature of the boric acid storage tank should not require monitoring as long as the room temperature is maintained ≥ 65°F. See paragraph 9.3.4.1.2.3 of the VEGP FSAR.

3.1.2.6:

The word "system" was replaced with the word "tank" for the reasons stated under 3.1.2.5.a, above.

The action statement shutdown margin requirement was revised to reflect the FSAR accident analysis results (boron dilution accident) which require a Shutdown Margin of (Later) delta - k/k at 200°F.

4.1.2.6.a.3:

See the justification provided for 4.1.2.5.a.3.

4.1.2.6.b:

The words "or greater than [100]°F" were deleted based on the maximum solution temperature of 120°F allowed for the RWST as discussed in paragraph 3.1.2.6.6.4 of the VEGP FSAR and, as discussed in FSAR paragraph 2.3.2.1.2, the extreme maximum outside air temperature in the 30-year period from 1941 to 1971 was 106°F.

3/4.1.3.1:

The generic STS specification was deleted in its entirety and a proposed version substituted in its place. The revision is a result of efforts made to address multiple immovable but trippable control rods. The generic STS technical specification does not recognize the fact that in this condition the control rods would still perform their safety function. The current STS requires that, in the case where a group or several groups of control rods became immovable due to a rod control system failure, the plant either repair the failure or be in hot standby in 6 hours. However, since a failure in the rod control system does not defeat the ability of the rods to trip, allowing several rods to be immovable but trippable is acceptable in that all analytical assumptions and requirements remain valid. This specification has been implemented at Diablo Canyon.

A revised bases is provided to support the proposed specification.

Table 3.1-1:

The titles of the accident analyses listed in this table were revised to be consistent with the VEGP FSAR. See subsections 15.0.5, 15.4.3, 15.6.1, 15.6.2, 15.6.5, 15.1.5, and 15.4.8 of the FSAR.

3.1.3.2:

The words "and shutdown" were added for clarification. Action Statement c was added to prevent confusion when returning either the digital rod position indication system or the demand position indication system to service after repair of the inoperable systems under Action 3.1.3.2.a.

4.1.3.3:

A surveillance interval was added to this surveillance which is consistent with recent NTOL submittals.

3/4.1.3.4:

The words "full length" were deleted on the basis that there are no part-length rods utilized in the VEGP design. Reference to N-1 loop operation was deleted reflecting the intent of the plant not to operate in N-1 loop operation. See paragraph 4.2.2.3 of the VEGP FSAR.

3.1.3.6:

The deletion of Figure 3.1-2 reflects the intent of the plant not to operate in N-1 loop operation.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION (N-0041, N-0042, N-0043, N-0044)

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the following target band (flux difference units) about the target flux difference:

a. ± 5% for core average accumulated burnup of less than or equal to 3000 MWD/MTU; and

b. + 3%, -12% for core average accumulated burnup of greater than 3000 MWD/MTU.

The indicated AFD may deviate outside the above required target band at greater than or equal to 50% but less than 90% of RATED THERMAL POWER provided the indicated AFD is within the Acceptable Operation Limits of Figure 3.2-1 and the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

The indicated AFD may deviate outside the above required target band at greater than 15% but less than 50% of RATED THERMAL POWER provided the cumulative penalty deviation time does not exceed 1 hour during the previous 24 hours.

APPLICABILITY: MODE 1, above 15% of RATED THERMAL POWER.*

ACTION:

a. With the indicated AFD outside of the above required target band and with THERMAL POWER greater than or equal to 90% of RATED THERMAL POWER, within 15 minutes either:

   1. Restore the indicated AFD to within the target band limits, or

   2. Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

b. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours or outside the Acceptable Operation Limits of Figure 3.2-1 and with THERMAL POWER less than 90% but equal to or greater than 50% of RATED THERMAL POWER, reduce:

   1. THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes, and

   2. The Power Range Neutron Flux* ** - High Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

---

*See Special Test Exceptions Specification 3.10.2.

**Surveillance testing of the Power Range Neutron Flux Channel may be performed pursuant to Specification 4.3.1.1 provided the indicated AFD is maintained within the Acceptable Operation Limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the above required target band during testing without penalty deviation.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band and the cumulative penalty deviation has been reduced to less than 1 hour in the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1) At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2) At least once per hour until ~~for the first 24 hours after restoring~~ the AFD Monitor Alarm to OPERABLE status is updated after restoration.

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its target band when two or more OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the above required target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each 1 minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and 0% at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER

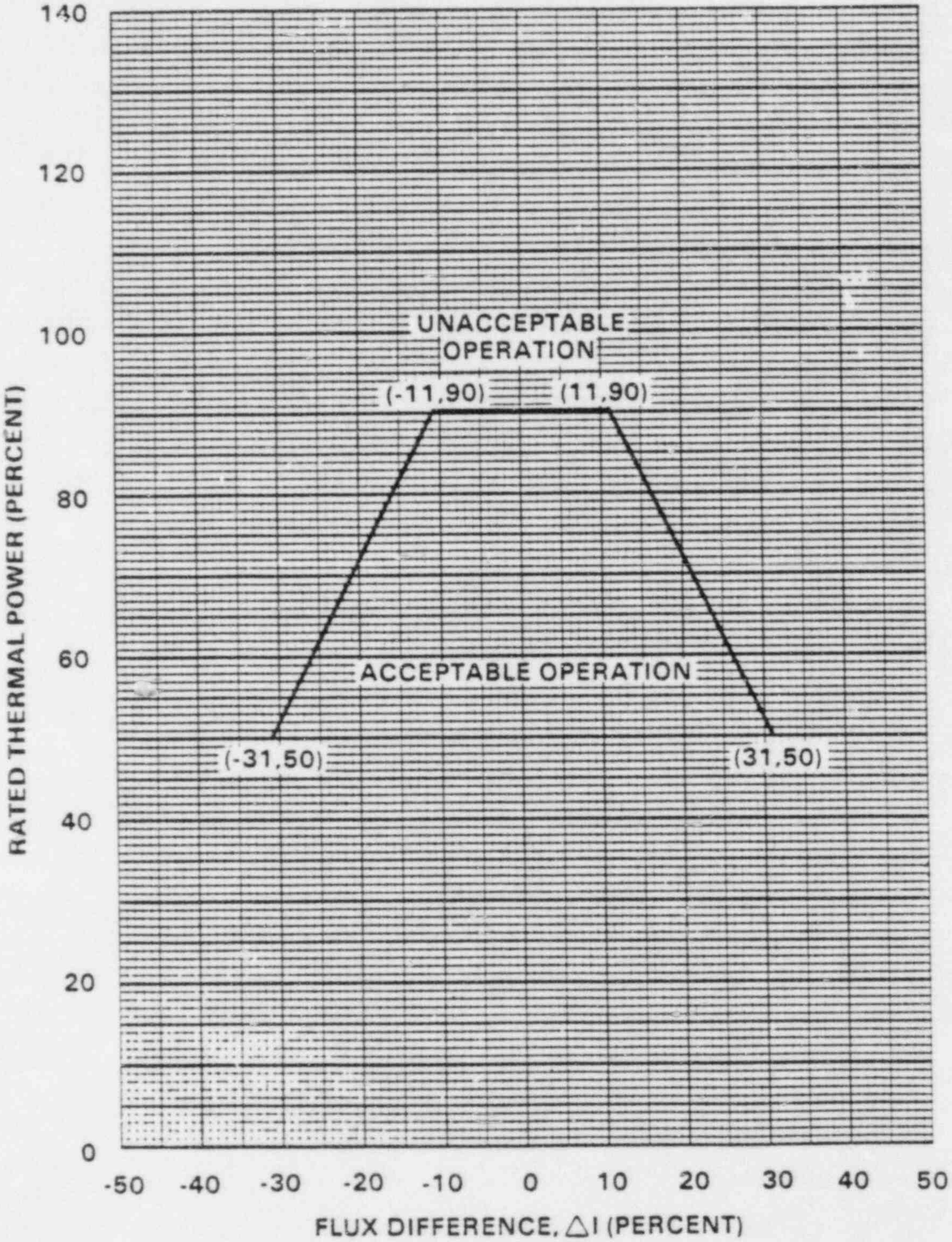


Figure 3.2-1. Axial Flux Difference Limits as a Function of Rated Thermal Power

POWER DISTRIBUTION LIMITS

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq \frac{\cancel{[2.32]}\ 2.30}{P} [K(Z)]$ for $P > 0.5$

$F_Q(Z) \leq \cancel{[(4.64)]}\ 4.60\ [K(Z)]$ for $P \leq 0.5$

Where: $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

K(Z) = the function obtained from Figure [3.2-2] for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

~~a. Comply with either of the following ACTIONS:~~

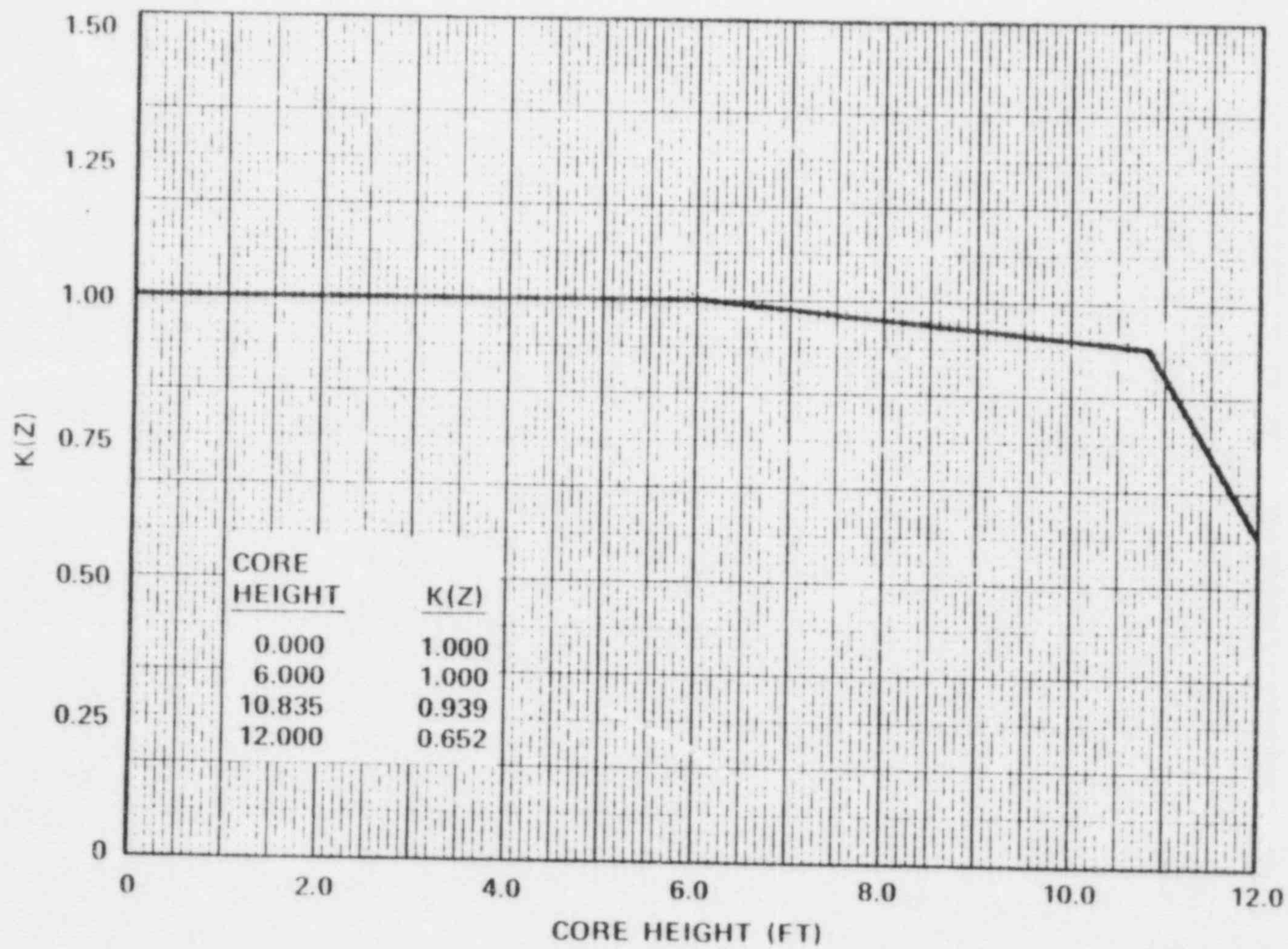
a ~~1.~~ Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit~~, or~~.

~~2. Reduce THERMAL POWER as necessary to meet the limits of Specification [3.2.6] using the Axial Power Distribution Monitoring Systems (APDMS) with the latest incore map and updated $\bar{R}$. [APDMS plants only]~~

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

FIGURE 3.2-2

K(Z) - NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT



CORE HEIGHT	K(Z)
0.000	1.000
6.000	1.000
10.835	0.939
12.000	0.652

FIGURE 3.2-2 K(Z) – NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F_{xy} shall be evaluated to determine if $F_Q(Z)$ is within its limit by:

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER, before exceeding 75% of RATED THERMAL POWER following each fuel loading.

b. Increasing the measured F_{xy} component of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties,

c. Comparing the F_{xy} computed (F_{xy}^{C}) obtained in Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core planes given in Specification 4.2.2.2e. and f., below, and

2) The relationship:

$$F_{xy}^{L} = F_{xy}^{RTP}\ [1+0.2(1-P)],$$

Where F_{xy}^{L} is the limit for fractional THERMAL POWER operation expressed as a function of F_{xy}^{RTP} and P is the fraction of RATED THERMAL POWER at which F_{xy} was measured.

d. Remeasuring F_{xy} according to the following schedule:

1) When F_{xy}^{C} is greater than the F_{xy}^{RTP} limit for the appropriate measured core plane but less than the F_{xy}^{L} relationship, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP}
½ space and F_{xy}^{L} either:

a) Within 24 hours after exceeding by 20% of RATED THERMAL POWER or greater, the THERMAL POWER at which F_{xy}^{C} was last determined, or

b) At least once per 31 Effective Full Power Days (EFPD), whichever occurs first.

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be provided for all core planes containing Bank "D" control rods and all unrodded core planes in a Radial Peaking Factor Limit Report per Specification 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2%, and 74.9 ± 2%, inclusive, and

4) Core plane regions within ± 2% of core height [± 2.88 inches] about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L},

~~1) The $F_Q(Z)$ limit shall be reduced at least 1% for each 1% F_{xy}^{C} exceeds F_{xy}, and (for plants with $F_Q(Z)$ less than 2.32 and using APDMS)~~

~~2)~~ The effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation shown on Figure 3.2-3 for four loop operation.

Where:

a. $R = \dfrac{F^N_{\Delta H}}{1.49\ [1.0 + 0.2\ (1.0 - P)]}$,

b. $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$, and

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by using the movable incore detectors to obtain a power distribution map. The measured values of $F^N_{\Delta H}$ shall be used to calculate R since Figure 3.2-3 includes penalties for undetected feedwater venturi fouling of [0.1]% and for measurement uncertainties of [2.1]% for flow and 4% for incore measurement of $F^N_{\Delta H}$.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation shown on Figure 3.2-3:

a. Within 2 hours either:

   1. Restore the combination of RCS total flow rate and R to within the above limits, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

3/4.2.3 NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

APPLICABILITY: MODE 1

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER TO LESS THAN 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours,

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

FIGURE 3.2-3

RCS TOTAL FLOW RATE VERSUS R - FOUR LOOPS IN OPERATION

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. Within 24 hours of initially being outside the above limits, verify through incore flux mapping and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through incore flux mapping and RCS total flow rate comparison, to be within the region of acceptable operation shown on Figure 3.2-3 prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

2. A nominal 75% of RATED THERMAL POWER, and

3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation of Figure 3.2-3:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation of Figure 3.2-3 at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

4.2.3.4 The RCS total flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to the performance of the calorimetric flow measurement.

4.2.3.5 The RCS total flow rate shall be determined by precision heat balance measurement at least once per 18 months.

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

POWER DISTRIBUTION LIMITS

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

(N-0041, N-0042, N-0043, N-0044)

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1, above 50% of RATED THERMAL POWER*.

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

    a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

    b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1, within 30 minutes;

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

    a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

    b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

3. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE, and

b. Calculating the ratio at least once per 12 hours during steady-state operation when the alarm is inoperable.

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from two sets of four symmetric thimble locations or full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}, ~~and~~ (TI-0412, TI-0422, TI-0432, TI-0442)

b. Pressurizer Pressure, (PI-0455 A, B, & C, PI-0456 A, B, & C, PI-0457A, B, & C, PI-0458A, B, & C)

c. Reactor Coolant System Total Flowrate

	Loop 1	Loop 2	Loop 3	Loop 4
	FT-0414	FT-0424	FT-0434	FT-0444
	FT-0415	FT-0425	FT-0435	FT-0445
	FT-0416	FT-0426	FT-0436	FT-0446

APPLICABILITY: MODE 1.*

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within its limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flowrate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months. The measurement instrumentation shall be calibrated within 7 days prior to performance of the calorimetric flow measurement.

4.2.5.3 The provisions of Specification 4.0.3 and 4.0.4 are not applicable for the first reactor startup following initial fuel load and each refueling for Reactor Coolant System total flowrate to allow the 18 month calorimetric flow measurement and the calibration of the Reactor Coolant System total flowrate indicators.

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* See Insert for page 3/4 2-13 for footnote.

Insert to Page 3/4 2-13

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* The provisions of Specification 3.0.2 are not applicable for the first reactor startup following initial core load and each refueling for Reactor Coolant System total flowrate to allow the 18-month calorimetric flow measurement and the calibration of the Reactor Coolant System total flowrate indicators.

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS N Loops in Operation	~~N-1 Loops in Operation & Loop Stop Valves Open~~	~~N-1 Loops in Operation & Loop Stop Valves Closed~~
Indicated Reactor Coolant System T_{avg}	$\le$ (Later) ~~[581]~~°F	~~$\le$ [569]°F~~	~~$\le$ [570]°F~~
Indicated Pressurizer Pressure	$\ge$ (Later) ~~[2220]~~ psi~~a~~g*	~~$\ge$ [2220] psia*~~	~~$\ge$ [2220]* psia~~
Indicated Reactor Coolant System Total Flow Rate	$\ge$ (Later) gpm		

*Limit not applicable during either a THERMAL POWER ramp in excess of ~~[~~5%~~]~~ of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of ~~[~~10~~]~~% of RATED THERMAL POWER.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.2

3/4.2:

Instrument tag numbers were added, as appropriate, to assist the operator in ensuring compliance with these Technical Specifications.

3.2.1, Action c:

This action statement has been revised in an effort to relieve a potential ambiguity. Without the revision, a literal interpretation of the action statement would allow power ascension above 50 percent immediately upon restoring the indicated AFD to within the target band without reducing the cumulative penalty deviation to less than 1 hour in the previous 24 hours. The intent of the specification has been preserved.

4.2.1.1.a.2:

The revision to this surveillance requirement meets the intent of ensuring a 24-hour history in the computer and it allows for controlled operator input to the computer.

3.2.2, Action a:

Deletion was made to reflect plant design (not an APDMS plant). See paragraphs 4.4.6.1 and 7.7.1.9 of the VEGP FSAR for a description of incore instrumentation.

4.2.2.2.a:

This change is more in line with the intent of the specification. As currently written in the generic STS the specification allows for operation without any surveillance of Fxy during the startup following each fuel loading.

4.2.2.2.g:

Deletion was made to reflect the VEGP design (not an APDMS plant).

3/4.2.3:

This specification has been deleted and replaced with a specification which defines the limit for the nuclear enthalpy hot channel factor ($F^N_{\Delta H}$) in terms of the design basis (1.55) and a factor which relaxes the limit on $F^N_{\Delta H}$ as a function of thermal power to allow for changes in the radial power shape for all permissible rod insertion limits. That is to say:

$$F^N_{\Delta H} \leq 1.55\ [1 + 0.3\ (1-P)]$$

$$\text{where}\quad P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$$

The design basis of 1.55 is used because the 4 percent allowance for experimental error will be applied to the measured valve of F as required by 4.2.3.2. See paragraph 4.4.4.3.1 of the VEGP FASR for a discussion of the above.

The limit for RCS flowrate has been moved to Specification 3/4.3.2.5.

Therefore, the revised specifications meet the intent of the orginal specification while providing simpler more straightforward requirements for the operation of the plant.

3/4.2.5:

The action statement has been changed to include the additional DNB parameter of reactor coolant system total flowrate as discussed in the justification for 3/4.2.3, above. This parameter was added due to the deletion of the rod bow penalty and its tradeoff with total flow in Specification 3.2.3.

Surveillance Requirement 4.2.5.2 was added to reflect the addition of RCS total flowrate to the LCO and is based on the generic STS Surveillance Requirement 4.2.3.5.

Footnote * and Surveillance Requirement 4.2.5.3 were added to allow the plant to enter Mode 1 to perform the calorimetric flow measurement and calibrate the reactor coolant system total flowrate indicators following the first reactor start-up after the initial core load and each refueling. This will allow the plant to demonstrate compliance with this specification's LCO upon completion of the calorimetric flow measurement and calibration of the reactor coolant system total flowrate indicators.

Table 3.2-1:

This change is a result of the addition of RCS total flowrate to specification 3/4.2.5 and the intent of the plant not to operate in N-1 loop operation.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*[a], 4*[a], 5*[a]	~~11~~ 9
2. Power Range, Neutron Flux (N0041, N0042, N0043, N0044)					
a. High Setpoint	4	2	3	1, 2	2~~#~~[b]
b. Low Setpoint	4	2	3	1###[d], 2	2~~#~~[b]
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2~~#~~[b]
(N0041, N0042, N0043, N0044)					
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2~~#~~[b]
(N0041, N0042, N0043, N0044)					
~~5. Intermediate Range, Neutron Flux~~	~~2~~	~~1~~	~~2~~	~~1###, 2~~	~~3~~
(N0035, N0036)					
6. Source Range, Neutron Flux (N0031, N0032)					
a. Startup	2	1	~~2~~ 1	2##[c][e]	4
b. Shutdown	2	1	2	3[b], 4, 5	5
7. Overtemperature ΔT	4	2	3	1, 2	6[b]
(TE-0411, TE-0421, TE-0431, TE-0441)					
~~a. Four Loop Plant~~					
~~Four Loop Operation~~	~~4~~	~~2~~	~~3~~	~~1, 2~~	~~6#~~
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
~~b. Three Loop Plant~~					
~~Three Loop Operation~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~
a. Power Operation	2	1	2	1[d]	3
b. Startup	2	1	1	2	3

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT	4	2	3	1, 2	6[b]
~~a. Four Loop Plant~~					
Four Loop Operation	4	2	3	1, 2	~~6#~~
~~Three Loop Operation~~	~~4~~	~~1**~~	~~3~~	~~1, 2~~	~~9~~
b. ~~Three Loop Plant~~					
~~Three Loop Operation~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
~~Two Loop Operation~~	~~3~~	~~1**~~	~~2~~	~~1, 2~~	~~9~~
(TE-0411, TE-0421, TE-0431, TE-0441)					
9. Pressurizer Pressure--Low	4	2	3	1[f] (See also Table 3.3-3)	6[b]
~~a. Four Loop Plant~~	~~4~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	~~3~~	~~2~~	~~2~~	~~1~~	~~7#~~
(PI-0455, PI-0456, PI-0457, PI-0458)					
10. Pressurizer Pressure--High	4	2	3	1, 2	6[b]
~~a. Four Loop Plant~~	~~4~~	~~2~~	~~3~~	~~1, 2~~	~~6#~~
~~b. Three Loop Plant~~	~~3~~	~~2~~	~~2~~	~~1, 2~~	~~7#~~
11. Pressurizer Water Level--High* (LI-0459, LI-0460, LI-0461)	3	2	2	1[f]	~~7#~~ 6[b]
12. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	~~7#~~ 6[b]
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	~~7#~~ 6[b]

Loop 1	Loop 2	Loop 3	Loop 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446

* See also Specification 3.3.3.6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Steam Generator Water Level--Low-Low *	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2 (See also Table 3.3-3)	6#[b, g]
~~14. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch~~	~~2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.~~	~~1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.~~	~~1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.~~	~~1, 2~~	~~7#~~
~~15.~~ 14 Undervoltage--Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1[f]	6[b]
~~a. Four Loop Plant~~	~~4-1/bus~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	~~3-1/bus~~	~~2~~	~~2~~	~~1~~	~~7#~~
~~16.~~ 15 Underfrequency--Reactor Coolant Pumps	4-2/bus	2-1/bus	3	1[f]	6[b]
~~a. Four Loop Plant~~	~~4-1/bus~~	~~2~~	~~3~~	~~1~~	~~6#~~
~~b. Three Loop Plant~~	~~3-1/bus~~	~~2~~	~~2~~	~~1~~	~~7#~~
~~17.~~ 16 Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1[e]	~~7#~~ 6[b]
b. Turbine Stop Valve Closure	4	4	1	1[e]	~~12#~~ 10[b]

(PI-6161, PI-6162, PI-6163)

Loop 1	Loop 2	Loop 3	Loop 4
LI-519	LI-518	LI-517	LI-551
LI-529	LI-528	LI-527	LI-552
LI-539	LI-538	LI-537	LI-553
LI-549	LI-548	LI-547	LI-554

* See also Specification 3.3.3.6.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
17 ~~18~~. Safety Injection Input from ESF	2	1	2	1, 2	~~10~~ 8
18 ~~19~~. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6 (N-0035, N-0036)	2	1	2	2##[c, e]	~~8~~ 7
b. Low Power Reactor Trips Block, P-7					
P-10 Input (N-0041, N-0042, N-0043, N-0044)	4	2	3	1	~~8~~ 7
or P-13 Input (PI-0505, PI-0506)	2	1	2	1	~~8~~ 7
c. Power Range Neutron Flux, P-8 (N-0041, N-0042, N-0043, N-0044)	4	2	3	1	~~8~~ 7
d. Power Range Neutron Flux, P-9 (N-0041, N-0042, N-0043, N-0044)	4	2	3	1	~~8~~ 7
e. Power Range Neutron Flux, P-10 (N-0041, N-0042, N-0043, N-0044)	4	2	3	1,2	~~8~~ 7
f. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	2	1	2	1	~~8~~ 7
19 ~~20~~. Reactor Trip Breakers	2	1	2	1, 2	~~10~~ 8, ~~13~~ 11
	2	1	2	3~~*~~[a], 4~~*~~[a], 5~~*~~[a]	~~11~~ 9
20 ~~21~~. Automatic Trip and Interlock Logic	2	1	2	1, 2	~~10~~ 8
	2	1	2	3~~*~~[a], 4~~*~~[a], 5~~*~~[a]	~~11~~ 9

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

a ~~*~~When the Reactor Trip System breakers are in the closed position and the Control Rod Drive System is capable of rod withdrawal.

~~**The channel(s) associated with the trip functions derived from the out-of-service reactor coolant loop shall be placed in the tripped condition.~~

b ~~#~~The provisions of Specification 3.0.4 are not applicable.

c ~~##~~Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

d ~~###~~Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

e Above the P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

f Above the P-7 (Low Power Reactor Trip Block) setpoint

g The Applicable Modes and Action statement for these channels noted in Table 3.3-3 are more restrictive and therefore, applicable.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~1~~ 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to ~~2~~ 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to [85]% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

h The Source Range High Flux at Shutdown Alarm may be blocked during reactor startup in accordance with approved procedures.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes.

Replace with insert to 3/4 3-7

ACTION 5 - ~~With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers, suspend all operations involving positive reactivity changes and verify Valves _______ are closed and secured in position within the next hour~~.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within ~~1~~ 6 hour~~s~~, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

~~ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.~~

ACTION ~~8~~ 7 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive ~~annunciator~~ window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

Insert to 3/4 3-7

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum channels OPERABLE requirements, restore the inoperable channels to OPERABLE status within 48 hours or open the Reactor Trip Breakers and suspend all operation involving positive reactivity charges and verify valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 are closed and secured in position within the next hour.

b. With no channels OPERABLE, open the Reactor Trip Breakers, suspend all operations involving positive reactivity charges, and verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and every 12 hours thereafter. Verify valves 1208-U4-175, 1208-U4-177, 1208-U4-183, and 1208-U4-176 are closed and secured in position within 4 hours and verified to be closed and secured in position every 14 days.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

~~ACTION 9 - With a channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1~~.

ACTION ~~10~~ 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION ~~11~~ 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION ~~12~~ 10 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within ~~1~~ 6 hours.

ACTION ~~13~~ 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 10. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	N.A.
2. Power Range, Neutron Flux (N-0041, N-0042, N-0043, N-0044)	$\leq$ [0.5] second*
3. Power Range, Neutron Flux, High Positive Rate (N-0041, N-0042, N-0043, N-0044)	N.A.
4. Power Range, Neutron Flux, High Negative Rate (N-0041, N-0042, N-0043, N-0044)	$\leq$ [0.5] second*
5. Intermediate Range, Neutron Flux (N-0035, N-0036)	N.A.
6. Source Range, Neutron Flux (N-0031, N-0032)	N.A.
7. Overtemperature ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	$\leq$ [4] seconds*
8. Overpower ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	$\leq$ [4] seconds*
9. Pressurizer Pressure--Low (PI-0455, PI-0456, PI-0457, PI-0458)	$\leq$ [2] seconds
10. Pressurizer Pressure--High (PI-0455, PI-0456, PI-0457, PI-0458)	$\leq$ [2] seconds
11. Pressurizer Water Level--High (LI-0459, LI-0460, LI-0461)	N.A.

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel. (This provision is not applicable to CPs docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)

Loop 1	Loop 2	Loop 3	Loop 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Reactor Coolant Flow--Low	
a. Single Loop (Above P-8)	$\leq$ [1] second
b. Two Loops (Above P-7 and below P-8)	$\leq$ [1] second
13. Steam Generator Water Level--Low-Low	$\leq$ [2] seconds
~~14. Steam Generator Water Level-Low Coincident with Steam/Feedwater Flow Mismatch~~	~~N.A.~~
14 15. Undervoltage - Reactor Coolant Pumps	$\leq$ [1.5] seconds
15 16. Underfrequency - Reactor Coolant Pumps	$\leq$ [0.6] second
16 17. Turbine Trip	
a. Low Fluid Oil Pressure (PI-6161, PI-6162, PI-6163)	N.A.
b. Turbine Stop Valve Closure	N.A.
17 18. Safety Injection Input from ESF	N.A.
18 19. Reactor Trip System Interlocks	N.A.
19 20. Reactor Trip Breakers	N.A.
20 21. Automatic Trip and Interlock Logic	N.A.

Loop 1	Loop 2	Loop 3	Loop 4
LI-0519	LI-0529	LI-0539	LI-0549
LI-0518	LI-0528	LI-0538	LI-0548
LI-0517	LI-0527	LI-0537	LI-0547
LI-0551	LI-0552	LI-0553	LI-0554

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3~~*~~[a], 4~~*~~[a], 5~~*~~[a]
2. Power Range, Neutron Flux (N-0041, N-0042, N-0043, N-0044)						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	~~M~~ Q(12)	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	~~M~~ S/U(12)	N.A.	N.A.	1~~***~~[d], 2
3. Power Range, Neutron Flux, High Positive Rate (N-0041, N-0042, N-0043, N-0044)	N.A.	R(4)	~~M~~ Q(12)	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate (N-0041, N-0042, N-0043, N-0044)	N.A.	R(4)	~~M~~ Q(12)	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux (N-0035, N-0036)	S	R(4, 5)	S/U(1), ~~M~~	N.A.	N.A.	1~~***~~[d], 2
6. Source Range, Neutron Flux (N-0031, N-0032)	S	R(4, 5, ~~12~~)	S/U(1), ~~M(9)~~ Q(8, 12)	N.A.	N.A.	2~~**~~[c], 3, 4, 5
7. Overtemperature ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	S	R(~~13~~ 11)	~~M~~ Q(12)	N.A.	N.A.	1, 2
8. Overpower ΔT (TE-0411, TE-0421, TE-0431, TE-0441)	S	R	~~M~~ Q(12)	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low (PI-0455, PI-0456, PI-0457, PI-0458)	S	R	~~M~~ Q(12)	N.A.	N.A.	1[e]
10. Pressurizer Pressure--High (PI-0455, PI-0456, PI-0457, PI-0458)	S	R	~~M~~ Q(12)	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High ✝ (LI-0459, LI-0460, LI-0461)	S	R	~~M~~ Q(12)	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	~~M~~ Q(12)	N.A.	N.A.	1

Loop 1	Loop 2	Loop 3	Loop 4
FI-0414	FI-0424	FI-0434	FI-0444
FI-0415	FI-0425	FI-0435	FI-0445
FI-0416	FI-0426	FI-0436	FI-0446

✝ See also Specification 4.3.3.6.

Loop 1	Loop 2	Loop 3	Loop 4
LI-0514	LI-0524	LI-0534	LI-0544
LI-0518	LI-0528	LI-0538	LI-0548
LI-0517	LI-0527	LI-0537	LI-0547
LI-0551	LI-0552	LI-0553	LI-0554

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level-- Low-Low	S	R	~~M~~ Q(12, 13)	N.A.	N.A.	1, 2
~~14. Steam Generator Water Level -- Low Coincident with Steam/ Feedwater Flow Mismatch~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~1, 2~~
14 ~~15~~. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q(12)	N.A.	1
15 ~~16~~. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	~~M~~ Q(12)	N.A.	1
16 ~~17~~. Turbine Trip						
a. Low Fluid Oil Pressure (PI-6161, PI-6162, PI-6163)	N.A.	R	~~N.A.~~ S/U(1, 9)	N.A ~~S/U(1, 10)~~	N.A.	1[b]
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, ~~10~~ 9)	N.A.	1[b]
17 ~~18~~. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
18 ~~19~~. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6 (N-0035, N-0036)	N.A.	R(4)	~~M~~ R	N.A.	N.A.	2[c] ~~2**~~
~~b. Low Power Reactor Trips Block, P-7~~	~~N.A.~~	~~R(4)~~	~~M(8)~~	~~N.A.~~	~~N.A.~~	~~1~~
b ~~c~~. Power Range Neutron Flux, P-8 (N-0041, N-0042, N-0043, N-0044)	N.A.	R(4)	~~M(8)~~ R	N.A.	N.A.	1
c ~~d~~. Power Range Neutron Flux, P-9 (N-0041, N-0042, N-0043, N-0044)	N.A.	R(4)	~~M(8)~~ R	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18 ~~20~~. Reactor Trip System Interlocks (Continued)						
d ~~e~~. Power Range Neutron Flux, P-10 (N-0041, N-0042, N-0043, N-0044)	N.A.	R(4)	R ~~M(8)~~	N.A.	N.A.	1, 2
c ~~f~~. Turbine Impulse Chamber Pressure, P-13 (PI-0505, PI-0506)	N.A.	R	R ~~M(8)~~	N.A.	N.A.	1
19 ~~21~~. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 10 ~~11~~)	N.A.	1, 2, 3~~*~~a, 4~~*~~a, 5~~*~~a
20 ~~22~~. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3~~*~~a, 4~~*~~a, 5~~*~~a
~~23. Reactor Trip Bypass Breaker~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~M(15),R(16)~~	~~N.A.~~	~~1, 2, 3*, 4*, 5*~~

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

a ~~*~~When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

b. Above P-9 (Reactor Trip on Turbine Trip Interlock) Setpoint.

c ~~**~~Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

d ~~***~~Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

e. Above P-7 (Low Power Reactor Trips Block) setpoint

(1) If not performed in previous 7 days.

(2) Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2%. The provisions of Specification 4.0.4 are not applicable to entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement monthly shall mean at least once per 31 EFPD.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained, and evaluated. ~~and compared to manufacturer's data~~. For the Intermediate Range and Power Range Neutron Flux channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. For the purpose of this surveillance requirement quarterly shall mean at least once per 92 EFPD.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

~~(8) With power greater than or equal to the Interlock Setpoint the required ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the interlock is in the required state by observing the permissive annunciator window.~~

8) Quarterly ~~(9) Monthly~~ surveillance in MODES 3~~*~~a, 4~~*~~a, and 5~~*~~a shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive ~~annunciator~~ window. Quarterly ~~Monthly~~ surveillance shall include verification that ~~of~~ the Source Range High Flux at Shutdown ~~Boron Dilution~~ Alarm Setpoint is set at ~~of~~ less than or equal to 3.16 times background. ~~(an increase of twice the count rate within a 10-minute period)~~.

Determination of the response of the excore power range detectors to the incore measured axial power distribution to generate setpoints for the channel calibration.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

9 ~~(10)~~ Setpoint verification is not applicable.

10 ~~(11)~~ At least once per 18 months and following maintenance or adjustment of the Reactor trip breakers, the TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips.

~~(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Doubling test signal the normal CVCS discharge valves will close and the centrifugal charging pumps suction valves from the RWST will open within [30] seconds~~.

(11 ~~13~~) CHANNEL CALIBRATION shall include the RTD bypass loops flow rate.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. ~~The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s)~~.

~~(15) Local manual shunt trip prior to placing breaker in service. (Or for plants that do not actuate the shunt trip attachment of the bypass breakers on a manual reactor trip): Remote manual undervoltage trip when breaker placed in service~~.

~~(16) Automatic undervoltage trip~~.

(12) Each channel shall be tested at least every 92 days on a Staggered Test Basis.

(13) The surveillance frequency and/or MODES specified for these channels in Table 4.3-2 are more restrictive and therefore applicable.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" column of Table 3.3-3.

VOGTLE - UNIT 1

3/4 3-18

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Component Cooling Water, Control Room ~~Isolation. Start~~ Emergency Mode Actuation, Start Diesel Generators, Containment Cooling Fans, ~~and Essential Service Water)~~. Nuclear Service Cooling Water, Containment Isolation Phase "A", Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps.)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	~~19~~ 17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
c. Containment Pressure--High-1# (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2, 3	~~15*~~ 18d
d. Pressurizer Pressure--Low (PI-0455, PI-0456, PI-0457, PI-0458)	4	2	3	1, 2, 3#	~~20*~~ 18d
~~e. Differential Pressure Between Steam Lines--High~~				~~1, 2, 3**~~	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15*~~
~~Three Loops Operating~~	~~3/operating steam line~~	~~1***/steam line any operating steam line~~	~~2/operating steam line~~		~~16~~

See also Specification 3.3.4

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (~~Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators~~, Containment ~~Cooling Fans, and Essential~~ Service Water) ~~(Continued)~~					
~~a) Three Loop Plant~~					
~~Three Loops Operating~~	3/steam line	~~2/steam line twice and 1/3 steam lines~~	~~2/steam line~~		~~15*~~
~~Two Loops Operating~~	3/operating steam line	~~2***/steam line twice in either operating steam line~~	~~2/operating steam line~~		~~16~~
e. ~~f.~~ Steam Line Pressure--low	4 3/steam line	2/steam line any steam line	2/steam line	1, 2, 3 a	18 d
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~1 pressure loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 3 loops~~		~~15*~~
~~Three Loops Operating~~	~~1 pressure/ operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure in in any 2 operating loops~~		~~16~~

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514	PI-0524	PI-0534	PI-0544
PI-0515	PI-0525	PI-0535	PI-0545
PI-0516	PI-0526	PI-0536	PI-0546

* See also Specification 3.3.3.6

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. ~~Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water) (Continued)~~					
2) ~~Three Loop Plant~~					
~~Three Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 2 loops~~		~~15*~~
~~Two Loops Operating~~	~~1 pressure/ loop~~	~~1*** pressure in any operating loop~~	~~1 pressure any operating loop~~		~~16~~
2. Containment Spray					
a. Manual Initiation	2	1 with 2 coincident switches	2	1, 2, 3, 4	~~19~~ 17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
c. Containment Pressure--High-3* (PI-0934, PI-0935, PI-0936, PI-0937)	4	2	3	1, 2, 3	~~17~~ 14

* See also Specification 3.3.3.4.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	2	1	2	1, 2, 3, 4	~~19~~ 17
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12
3) Safety Injection	See ~~Item~~ Functional Unit 1. above for all Safety Injection initiating functions and requirements.				
4) Containment Area Radiation (High Range)# (RE-0005, RE-0006)	2	1	2	1, 2, 3, 4	13
~~b. Phase "B" Isolation~~					
~~1) Manual Initiation~~	~~2~~	~~1 with 2 coincident switches~~	~~2~~	~~1, 2, 3, 4~~	~~19~~
~~2) Automatic Actuation Logic and Actuation Relays~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4~~	~~14~~
~~3) Containment Pressure-High-3~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~17~~
b. Containment Ventilation Isolation ~~c. Purge and Exhaust Isolation~~					
1) Manual Initiation	2	1	2	1, 2, 3, 4, 6[c]	~~18~~ 15, 16
2) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4, 6[c]	~~18~~ 15, 16
3) Safety Injection	See ~~Item 1~~ Functional Unit. above for all Safety Injection initiating functions and requirements.				

See also Specification 3.3.3.6.

See insert for 3/4 3-21.

Insert for 3/4 3-21

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3b. Continued					
4. Containment Area Radiation (Low Range) (RE-0002, RE-0003)	2	1	2	1, 2, 3, 4, 6^c	15, 16
5. Containment Ventilation Radiation					
i) Particulate (RE-2565A)	1	1	9	1, 2, 3, 4, 6^c	15, 16
ii) Iodine (RE-2565B)	1	1	9	1, 2, 3, 4, 6^c	15, 16
iii) Gaseous (RE-2565C)	1	1	9	1, 2, 3, 4, 6^c	15, 16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	~~1~~ 2/steam line	1/steam line	1/Operating steam line	1, 2[f], 3[f]	~~24~~ 22
2) System	2	1	2	1, 2[f], 3[f]	~~23~~ 21
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2[f], 3[f]	~~22~~ 20
c. Containment Pressure--High-2 ‡ (PI-0934, PI-0935, PI-0936)	3	2	2	1, 2[f], 3[f]	15* 18[d]
~~d. Steam Flow in Two Steam Lines--High~~				~~1, 2, 3~~	
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Three Loops Operating~~	~~2/operating steam line~~	~~1###/any operating steam line~~	~~1/operating steam line~~		~~16~~
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~2/steam line~~	~~1/steam line any 2 steam lines~~	~~1/steam line~~		~~15*~~
~~Two Loops Operating~~	~~2/operating steam line~~	~~1###/any operating steam line~~	~~1/operating steam line~~		~~16~~

‡ See also Specification 3.3.3.1.

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514	PI-0524	PI-0534	PI-0544
PI-0515	PI-0525	PI-0535	PI-0545
PI-0516	PI-0526	PI-0536	PI-0546

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d ~~e~~. Steam Line Pressure--Low *	3/steam line	2/steam line any steam line	2/steam line	1, 2[f], 3[a,f] ~~1, 2, 3**~~	18[d]
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 3 loops~~		~~15*~~
~~Three Loops Operating~~	~~1 pressure/ operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure in any 2 operating loops~~		~~16~~
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~1 pressure/ loop~~	~~1 pressure any 2 loops~~	~~1 pressure any 2 loops~~		~~15*~~
~~Two Loops Operating~~	~~1 pressure/ operating loop~~	~~1*** pressure in any operating loop~~	~~1 pressure any operating loop~~		~~16~~
e ~~f~~. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/steam line	~~3****~~ 3[b,f]	18[d]
~~1) Four Loop Plant~~					
~~Four Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~15*~~
~~Three Loops Operating~~	~~3/operating steam line~~	~~2/steam line in any operating steam line~~	~~2/steam line in each operating steam line~~		~~16~~

* See also Specification 3.3.3.6.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
Steam Flow in Two Steam Lines--High					
Coincident With:					
T_{avg}--Low-Low				1, 2, 3	
1) Four Loop Plant					
Four Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 3 loops		15*
Three Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any two operating loops		16
2) Three Loop Plant					
Three Loops Operating	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops		15*
Two Loops Operating	1 T_{avg}/operating loop	1*** T_{avg} in any operating loop	1 T_{avg} in any operating loop		16

Loop 1	Loop 2	Loop 3	Loop 4
LI-0519	LI-0529	LI-0539	LI-0549
LI-0518	LI-0528	LI-0538	LI-0548
LI-0517	LI-0527	LI-0537	LI-0547
LI-0551	LI-0552	LI-0553	LI-0554

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~4.~~ Steam Line Isolation (Continued)					
~~2) Three Loop Plant~~					
~~Three Loops Operating~~	~~3/steam line~~	~~2/steam line any steam line~~	~~2/steam line~~		~~16~~
~~Two Loops Operating~~	~~3/operating steam line~~	~~2/steam line in any operating steam line~~	~~2/steam line in each operating steam line~~		~~16~~
5. Turbine Trip and Feedwater Isolation					
a. Safety Injection → See Functional Unit 1 above, for all Safety Injection initiating functions and requirements.					
b. ~~a.~~ Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	~~25~~ 23
c. ~~b.~~ Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2	~~20*~~ 18d
6. Auxiliary Feedwater					
~~a. Manual Initiation~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3~~	~~23~~
a. ~~b.~~ Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	~~22~~ 20
b. ~~c.~~ Stm. Gen. Water Level--Low-Low*					
1) Start Motor-Driven Pumps	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	~~20*~~ 18d

* See also Specification 3.3.3.6.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. Auxiliary Feedwater (Continued)					
2) Start Turbine-Driven Pump	4/stm. gen.	2/stm. gen. in any 2 operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	~~20*~~ 18[d]
~~d. Undervoltage RCP Start Turbine-Driven Pump~~	~~4-1/bus~~	~~2~~	~~3~~	~~1, 2~~	~~20*~~
c. ~~e.~~ Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection initiating functions and requirements.				
d. ~~f.~~ Loss-of-~~Offsite Power~~ or Degraded 4.16 kV ESF Bus Voltage					
i Start Motor-Driven	4/train	2/train	3/train	1, 2, 3	21
ii Pumps ~~and~~, Turbine-Driven Pump	8 ~~2~~	~~1~~ 2/from either train	~~2~~ 3/train	1, 2, 3	~~19~~ 21
e. ~~g.~~ Trip of All Main Feedwater Pumps Start Motor-Driven Pumps and Turbine-Driven Pump	2/~~pump~~	2 ~~1/pump~~	2 ~~1/pump~~	1, 2	~~19~~ 25[d]
~~h. Suction Transfer on Low Pressure~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~19~~
7. Semi-Automatic Switchover to Containment Emergency Sump					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	~~14~~ 12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. Semi-Automatic Switchover to Containment Sump (Continued)					
b. RWST Emergency Level--Low-Low	4	2	3	1, 2, 3, 4	~~17~~ 14
Coincident with Safety Injection ~~Coincident With~~:					
~~Containment Sump Level--High~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3, 4~~	~~17~~
~~and~~					
~~Safety Injection~~	See Item 1. above for all Safety Injection initiating functions and requirements.				
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 ~~4~~ kV Emergency Bus Undervoltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	~~20*~~ 18d
b. 4.16 ~~4~~ kV Emergency Bus Undervoltage-~~Grid~~ Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	~~20*~~ 18d
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455, PI-0456, PI-0457)	3	2	2	1, 2, 3	~~21~~ 19
~~b. Low-Low~~ ~~T_{avg}~~~~, P-12~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~21~~
b. ~~c.~~ Reactor Trip, P-4	2	~~2~~ 1	2	1, 2, 3	~~23~~ 21
~~d. Steam Generator Water Level, P-14~~	~~3/stm. gen.~~	~~2/stm. gen. in any operating stm. gen.~~	~~2/stm. gen. in each operating stm. gen.~~	~~1, 2, 3~~	~~21~~

See insert for 3/4 3-26.

INSERT TO PAGE 3/4 3-26 (TABLE 3.3-3)

10. CONTROL ROOM VENTILATION EMERGENCY MODE ACTUATION					
a. Manual Initiation	2	1	1	1, 2, 3, 4, 5[e], 6[e]	24[d]
b. Automatic Actuation Logic and Actuation Relays	2	1	1	1, 2, 3, 4, 5[e], 6[e]	24[d]
c. Safety Injection	See FUNCTIONAL UNIT 1 above for all Safety Injection initiating functions and requirements				
d. Intake Radiogas Monitor (RE-12116, RE-12117)	2	1	1	1, 2, 3, 4, 5[e], 6[e]	24[d]

e. During movement of irradiated fuel or movement of loads over irradiated fuel.
f. Not applicable if the main steam isolation valves are closed.
g. At least two Containment Ventilation Radiation instruments must be OPERABLE.

TABLE 3.3-3 (Continued)

TABLE NOTATIONS

d ~~*~~The provisions of Specification 3.0.4 are not applicable.

a ~~#~~Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

c During movement of irradiated fuel or movement of loads over irradiated fuel within containment.
~~**Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) Setpoint.~~

~~***The channel(s) associated with the protective functions derived from the out-of-service reactor coolant loop shall be placed in the tripped mode.~~

b ~~****~~Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

ACTION STATEMENTS

ACTION ~~14~~ 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

See insert for 3/4 3-27

~~ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.~~

ACTION 16 - With a ~~channel associated with an operating loop inoperable, restore the inoperable channel to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours. One channel associated with an operating loop may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1~~.

ACTION ~~17~~ 14 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION ~~18~~ 15 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are ~~maintained~~ closed within 24 hours (Modes 1,2,3,4).

Insert for 3/4 3-27

ACTION 13 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, restore the number of OPERABLE channels to the Minimum Channel OPERABLE requirement within 7 days, or be in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN within the subsequent 24 hours.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 16 - See insert to 3/4 3-28.

ACTION ~~19~~ 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION ~~20~~ 18 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirements ~~is~~ are met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION ~~21~~ 19 - With the number of OPERABLE channels less than the Minimum ~~Number of~~ Channels OPERABLE requirement, within 1 hour determine by observation of the associated permissive ~~annunciator~~ window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION ~~22~~ 20 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION ~~23~~ 21 - With the number of OPERABLE channels one less than the ~~Total Number of~~ Minimum Channels OPERABLE, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION ~~24~~ 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification ~~{~~3.7.1.5~~}~~.

ACTION ~~25~~ 23 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 24 }
ACTION 25 } See insert for 3/4 3-28.

Insert to page 3/4 3-28

ACTION 16 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9 (Mode 6).

ACTION 24 - With the number of OPERABLE channels one less than the total number of channels, resto e the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Ventilation System in the Emergency mode. With two channels inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the Emergency mode.

ACTION 25 - with the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, operation may proceed until the performance of the next required TRIP ACTUATING DEVICE OPERATIONAL TEST.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control ~~Room Isolation~~ Room Emergency Mode Actuation, Component Cooling Water, Start Diesel Generators, Containment Cooling Fans, ~~and Essential~~ Nuclear Service Water Cooling, Containment Isolation Phase "A," Containment Ventilation Isolation, Auxiliary Feedwater Motor-Driven Pumps.)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High 1 (PI-0934, PI-0935, PI-0936, PI-0937)	~~[~~3.0~~]~~	~~[~~0.71~~]~~	~~[~~1.5~~]~~	≤ ~~[3.6]~~ 4.0 psig	≤ ~~[3.86]~~ 4.8 psig
d. Pressurizer Pressure--Low (PI-0455, PI-0456, PI-0457, PI-0458)	~~[~~13.1~~]~~	~~[~~10.71~~]~~	~~[~~1.5~~]~~	≤ ~~[~~1850~~]~~ psig	≤ ~~[~~183~~9]~~ 8 psig
~~e. Differential Pressure Between Steam Lines--High~~	~~[3.0]~~	~~[0.87]~~	~~[1.5/ 1.5]~~	~~≤ [97] psi~~	~~≤ [106] psi~~
e. ~~f.~~ Steam Line Pressure--Low	~~[20.0]~~ 13.1	~~[~~10.71~~]~~	~~[~~1.5~~]~~	≤ ~~[675]~~ 585 psig	≤ ~~[635]~~ 567 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3 (PI-0934, PI-0935, PI-0936, PI-0937)	~~[~~3.0~~]~~	~~[~~0.71~~]~~	~~[~~1.5~~]~~	≤ ~~[12.05]~~ 22 psig	≤ ~~[12.31]~~ 22.8 psig

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514 PI-0515 PI-0516	PI-0524 PI-0525 PI-0526	PI-0534 PI-0535 PI-0536	PI-0544 PI-0545 PI-0546

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
4) Containment Area Radiation (High Range) (RE-0005, RE-0006)			LATER		
~~b. Phase "B" Isolation~~					
~~1) Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
~~2) Automatic Actuation Logic and Actuation Relays~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
~~3) Containment Pressure--High-3~~	~~[3.0]~~	~~[0.71]~~	~~[1.5]~~	~~≤ [12.05] psig~~	~~≤ [12.31] psig~~
b. ~~c.~~ Containment Ventilation ~~Purge and Exhaust~~ Isolation					
~~1) Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
1) ~~2)~~ Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
2) ~~3)~~ Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

See insert for 3/4 3-30

Insert for 3/4 3-30

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
2.b continued					
3) Containment Area Radioactivity (Low Range) (RE-0002, RE-0003)	Later		Later	Later	Later
4) Containment Ventilation Radiation					
(i) Particulate Activity (RE-2565A)	Later		Later	Later	Later
(ii) Iodine Activity (RE-2565B)	Later		Later	Later	Later
(iii) Gaseous Activity (RE-2565C)	Later		Later	Later	Later

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-2	[3.0]	[0.71]	[1.5]	≤ 15 ~~[≤6.35]~~ psig	≤ 15.8 ~~[≤6.61]~~ psig
~~d. Steam Flow in Two Steam Lines-- High, Coincident with~~	~~[20.0]~~	~~[13.16]~~	~~[1.5/ 1.5]~~	~~≤ A function defined as follows: A ΔP corresponding to 40% of full steam flow between 0% and 20% load and then a ΔP increasing linearly to 110% of full load.~~	~~≤ A function defined as follows: A ΔP corresponding to 44% of full steam flow between 0% and load and then a ΔP increasing linearly to a ΔP corresponding to 114.0% of full steam flow at full load.~~
~~T_{avg}--Low-Low~~	~~[4.0]~~	~~[1.12]~~	~~[1.2]~~	~~≤[553]°F~~	~~≤[550.6]°F~~
d. ~~e.~~ Steam Line Pressure--Low	13.1 ~~[20.0]~~	[10.71]	[1.5]	≤ 585 ~~[675]~~ psig	≤ 567 ~~[635]~~ psig*
e. ~~f.~~ Steam Line Pressure - Negative Rate--High	5.0 ~~[8.0]~~	[0.5]	[0]	≤ -100 ~~[110]~~ psi/s	≤ -123 ~~[121.6]~~ psi/s**

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514	PI-0524	PI-0534	PI-0544
PI-0515	PI-0525	PI-0535	PI-0545
PI-0516	PI-0526	PI-0536	PI-0546

Loop 1	Loop 2	Loop 3	Loop 4
LI-0517	LI-0527	LI-0537	LI-0547
LI-0518	LI-0528	LI-0538	LI-0548
LI-0519	LI-0529	LI-0539	LI-0549
LI-0551	LI-0552	LI-0553	LI-0554

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	5.1 ~~[5.0]~~	[2.18]	[1.5]	<~~[82.4]~~ 78% of narrow range instrument span.	<~~[84.2]~~ 79.8% of narrow range instrument span.
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~
a. ~~b.~~ Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. ~~c.~~ Steam Generator Water Level--Low-Low	17.0 ~~[30.0]~~	12.18 ~~[27.18]~~	[1.5]	> ~~[32.2]~~ 17% of narrow range instrument span.	> ~~[30.4]~~ 15.3% of narrow range instrument span.
1) Start Motor-Driven Pumps					
~~d. Undervoltage - RCP~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~< [70]% RCP bus voltage.~~	~~< [69]% RCP bus voltage.~~
2) Start Turbine-Driven Pumps	17.0	12.18	1.5	≥ 17% of narrow range instrument span.	≥ 15.3% of narrow range instrument span.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
6. Auxiliary Feedwater (Continued)					
c ~~e.~~ Safety Injection Start Motor-Driven Pumps	See Functional Unit ~~Item~~ 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
d ~~f.~~ Loss-of-~~Offsite Power~~ or Degraded 4.16 kV ESF Bus Voltage Start Motor-Driven and Turbine-Driven Pumps	N.A.	N.A.	N.A.	3598 volts with 20 s time delay ~~≥ [4800]V~~ *	≥ 3548 volts with ≤ 20 s time delay ~~≥ [4692]V~~
e ~~g.~~ Trip of All Main Feedwater Pumps, Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
~~h. Suction Transfer on Low Pressure~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [442] ft~~	~~≤ [441] ft~~
7. Semi-Automatic Switchover to Containment Emergency Sump					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level--Low-Low ~~Coincident With~~	N.A.	N.A.	N.A.	274 in. from tank base ~~≤ [18]%~~ 35.6 % of instrument span	254 in. from tank base ~~≤ [15]%~~ 32.5 % of instrument span
~~Containment Sump Level~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≤ [30] in. above [680] ft~~	~~≤ [32.5] in. above [680) ft~~
Coincident With ~~and~~ Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power to 4.16 kV ESF Bus					
a. 4.16 ~~4~~ kV Emergency Bus Undervoltage ~~(Loss of Voltage)~~	N.A.	N.A.	N.A.	2912 ~~≤ [5760]~~ volts with a ≤ ~~[0.25]~~ 0.8 second time delay.	≥ 2912 ~~≤ [5652]~~ volts with a ≤ ~~[0.275]~~ 0.8 second time delay.
b. 4.16 ~~4~~ kV Emergency Bus Undervoltage (Grid Degraded Voltage)	N.A.	N.A.	N.A.	3598 ~~≤ [6576]~~ volts with a ~~≤ [3.3]~~ 20 second time delay.	≥ 3598 ~~≤ [6511]~~ volts with a ~~≤ [3.3]~~ 20 second time delay.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11 (PI-0455, PI-0456, PI-0457, PI-0458)	N.A.	N.A.	N.A.	≤ ~~[1985]~~ 1970 psig	≤ ~~[1996]~~ 1981 psig
~~b. Low-Low T_{avg}, P-12~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~≥ [553]°F~~	~~≤ [550.6]°F &~~
b ~~c.~~ Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
~~d. Steam Generator Water Level, P-14~~	~~See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.~~				
10. Control Room Ventilation Emergency Mode Activation					
a. Manual Initiation	N.A	N.A.	N.A.	N.A	N.A
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A	N.A.
c. Safety Injection	See Functional Unit 1 above for all Safety Injection Trip Setpoints/ Allowable Values				
d. Intake Radiogas Monitor (RE 12116, 12117)	(Later)	(Later)	(Later)	(Later)	(Later)

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq$ [50] seconds and $\tau_2 \geq$ [5] seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is less than or equal to [50] seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATION SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
~~d. Phase "B" Isolation~~	~~N.A.~~
~~e. Purge and Exhaust Isolation~~	~~N.A.~~
d. ~~f.~~ Steam Line Isolation	N.A.
~~g. Feedwater Isolation~~	~~N.A.~~
~~h. Auxiliary Feedwater~~	~~N.A.~~
~~i. Essential Service Water~~	~~N.A.~~
~~j. Containment Cooling Fans~~	~~N.A.~~
e. ~~k.~~ Control Room ~~Isolation~~ Ventilation Emergency Mode Actuation	N.A.
f. ~~l.~~ Reactor Trip	N.A.
g. ~~m.~~ Start Diesel Generator	N.A.
2. Containment Pressure--High-1	
a. Safety Injection (ECCS)	≤ ~~[27]~~ 29.0 (1) ~~(1)~~ / [12] (5)
b. ~~1)~~ Reactor Trip (from SI)	≤ ~~[2]~~ 2.0
c. ~~2)~~ Feedwater Isolation	≤ ~~[7]~~ 7.0 (3)
d. ~~3)~~ Phase "A" Isolation	≤ ~~[17] (2) / [27] (1)~~ 2.0 (6)
e. ~~4)~~ ~~Purge and Exhaust~~ Containment Ventilation Isolation	≤ ~~[25] (1) / [10] (2)~~ 6.5
f. ~~5)~~ Auxiliary Feedwater	≤ ~~[60]~~ 60.0
g. ~~6)~~ ~~Essential Service~~ Nuclear Service and Component Cooling Water	≤ ~~[32] (1) / [47] (2)~~ 100.0 (1) / 86.0 (2)
h. ~~7)~~ Containment Cooling Fans	≤ ~~[55] (1) / [40] (2)~~ 40.0 (1) / 28.0
i. ~~8)~~ Control Room ~~Isolation~~ Ventilation Emergency Mode Actuation	N.A.
~~9) Start Diesel Generator~~	~~≤ [10]~~

Feedwater Isolation	N.A.
Component Cooling Water	N.A.
Containment Cooling Fans	N.A.
Nuclear Service Cooling Water	N.A.
Auxiliary Feedwater Motor-Driven Pumps	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	≤ ~~[27]~~ 29.0(1)/~~[12]~~ 12.0(5)
b. ~~1)~~ Reactor Trip (from SI)	≤ ~~[2]~~ 2.0
c. ~~2)~~ Feedwater Isolation	≤ ~~[7]~~ 7.0(3)
d. ~~3)~~ Phase "A" Isolation	≤ ~~[17](2)/[27](1)~~ 2.0(6)
e. ~~4)~~ Containment Ventilation ~~Purge and Exhaust~~ Isolation	≤ ~~[25](1)/[10](2)~~ 6.5
f. ~~5)~~ Auxiliary Feedwater	≤ ~~[60]~~
g. ~~6)~~ Nuclear Service and Component Cooling ~~Essential Service~~ Water	≤ ~~[47]~~ 100.0(1)/~~[32]~~ 86.0(2)
h. ~~7)~~ Containment Cooling Fans	≤ ~~[55]~~ 40.0(1)/~~[40]~~ 28.0(2)
i. ~~8)~~ Control Room ~~Isolation~~ Ventilation Emergency Mode Actuation	N.A.
~~9) Start Diesel Generators~~	~~≤ [10]~~
~~4. Differential Pressure Between Steam Lines--High~~	
~~a. Safety Injection (ECCS)~~	~~≤ [22](4)/[12](5)~~
~~1) Reactor Trip~~	~~≤ [2]~~
~~2) Feedwater Isolation~~	~~≤ [7](3)~~
~~3) Phase "A" Isolation~~	~~≤ [17](2)/[27](1)~~
~~4) Purge and Exhaust Isolation~~	~~≤ [25](1)/[10](2)~~
~~5) Auxiliary Feedwater~~	~~≤ [60]~~
~~6) Essential Service Water~~	~~≤ [32](2)/[47](1)~~
~~7) Containment Cooling Fans~~	~~≤ [55](1)/[40](2)~~
~~8) Control Room Isolation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~≤ [10]~~
4 ~~5.~~ Steam Line Pressure--Low	
a. Safety Injection (ECCS)	≤ [12](5)/~~[22]~~ 24(4)
b. ~~1)~~ Reactor Trip (from SI)	≤ ~~[2]~~
c. ~~2)~~ Feedwater Isolation	≤ ~~[7]~~(3)
d. ~~3)~~ Phase "A" Isolation	≤ ~~[17](2)/[27](1)~~ 2.0(6)
e. ~~4)~~ Containment Ventilation ~~Purge and Exhaust~~ Isolation	≤ ~~[25](1)/[10](2)~~ 6.5
f. ~~5)~~ Auxiliary Feedwater	≤ ~~[60]~~ 60.0
g. ~~6)~~ Nuclear Service and Component Cooling ~~Essential Service~~ Water	≤ ~~[32](2)/[47](1)~~ 100.0(1)/86.0(2)
h. ~~7)~~ Containment Cooling Fans	≤ ~~[55]~~ 40.0(1)/~~[40]~~ 28.0(2)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4 ~~5.~~ Steam Line Pressure--Low (Continued)	
i. ~~8)~~ Control Room, ~~Isolation~~ Ventilation Emergency Mode Actuation	N.A.
j. ~~9) Start Diesel Generators~~	~~$\leq$ [10]~~
b. Steam Line Isolation	$\leq$ [9]$^{(3)}$
5 ~~6.~~ Containment Pressure--High-3	
a. Containment Spray	$\leq$ ~~[45]~~ 88.0$^{(2)}$/~~[57]~~ 100.0$^{(1)}$
~~b. Phase "B" Isolation~~	~~$\leq$ [65]$^{(1)}$/[75]$^{(2)}$~~
6 ~~7.~~ Containment Pressure--High-2	
a. Steam Line Isolation	$\leq$ ~~[9]~~ 7.0$^{(3)}$
~~8. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low~~	
~~Steam Line Isolation~~	~~$\leq$ [9]$^{(3)}$~~
7 ~~9.~~ Steam Line Pressure - Negative Rate--High	
a. Steam Line Isolation	$\leq$ ~~[9]~~ 7.0$^{(3)}$
8 ~~10.~~ Steam Generator Water Level--High-High	
a. Turbine Trip	$\leq$ [2.5]
b. Feedwater Isolation	$\leq$ ~~[7]$^{(3)}$~~ 7.0$^{(3)}$
9 ~~11.~~ Steam Generator Water Level--Low-Low	
a. Motor-Driven Auxiliary Feedwater Pumps	$\leq$ [60]
b. Turbine-Driven Auxiliary Feedwater Pump	$\leq$ [60]
~~12. Undervoltage RCP~~	
~~Turbine-Driven Auxiliary Feedwater Pump~~	~~$\leq$ [60]~~
10 ~~13.~~ Loss-of-~~Offsite Power~~ or Degraded 4.16 kV ESF Bus Voltage	
a. ~~Turbine-Driven~~ Auxiliary Feedwater ~~Pump~~	$\leq$ [60]
11 ~~14.~~ Trip of ~~All~~ Main Feedwater Pumps	
a. ~~All~~ Auxiliary Feedwater ~~Pumps~~	N.A.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
~~15. Suction Transfer on Low Pressure~~	
~~Auxiliary Feedwater (Suction Supply Automatic Realignment)~~	~~$\le$ [13]~~
12 ~~16.~~ RWST Level--Low-Low Coincident with Safety Injection	
~~a. Automatic Switchover to Containment Sump~~	~~N.A.~~
~~b. Coincident with Containment Sump Level--High and Safety Injection~~	
Semi-~~(~~Automatic Switchover to Containment Emergency Sump~~)~~	$\le$ ~~[250]~~ 46.0(2)/~~[265]~~ 60.0(1)
13 ~~17.~~ Loss of Power	
a. ~~4~~ 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	~~$\le$ [10]~~
Start signal to Diesel Generator	$\le$ 2.0 *
b. ~~4~~ 4.16 kV ~~Emergency~~ ESF Bus Undervoltage (Grid Degraded Voltage)	~~$\le$ [10]~~
Start signal to Diesel Generater	$\le$ 21.2 **
14. Control Room Intake Radiogas	
a. Control Room Ventilation Emergency Mode Actuation	$\le$ 10.0 (Isolation)
15. Containment Radioactivity	
a. Area Radiation Low Range - Containment Ventilation Isolation	$\le$ 15.0
b. Containment Ventilation Radiation - Containment Ventilation Isolation	$\le$ 15.0
c. Area Radiation High Range - Containment Isolation Phase A	$\le$ 15.0

TABLE 3.3-5 (Continued)

TABLE NOTATIONS

(1) Signal sensing Diesel generator starting and sequence loading delays included.

(2) Signal sensing Diesel generator starting and sequence loading delay <u>not</u> included. Offsite power available.

(3) Electrohydraulic ~~Air-operated~~ valves.

(4) Signal sensing Diesel generator starting and sequence loading delay included. RHR pumps <u>not</u> included.

(5) Signal sensing Diesel generator starting and sequence loading delays not included. RHR pumps <u>not</u> included.

(6) Response time includes signal conditioning and processing times only

* The response time shall include the time delay associated with the loss of voltage relays as determined in Table 3.3-4 plus an additional 1.2 seconds associated with interposing relay and circuit operation.

** The response time shall include the time delay associated with the undervoltage relays as determined in Table 3.3-4 plus an additional 1.2 seconds associated with interposing relay and circuit operation.

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514	PI-0524	PI-0534	PI-0544
PI-0515	PI-0525	PI-0535	PI-0545
PI-0516	PI-0526	PI-0536	PI-0546

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, ~~Control~~ Component Cooling Water Control Room Isolation, ~~Start Diesel~~ Emergency Mode Actuation Start Diesel Generators, Containment Cooling Fans, ~~and Essential Service Water)~~ Nuclear Service Cooling Water, Containment Isolation Phase A, Containment Ventilation Isolation, and Auxiliary Feedwater Motor-Driven Pumps)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-1+ (PI-0934, PI-0935, PI-0936)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low (PI-0455, PI-0456, PI-0457, PI-0458)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~e. Differential Pressure Between Steam Lines-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
e. ~~f.~~ Steam Line Pressure-Low+	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-High-3* (PI-0934, PI-0935, PI-0936, PI-0937)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

+ See also Specification 4.3.3.6

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	~~See Item~~ Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
4) Containment Area Radiation (High Range)* (RE-0005, RE-0006)	S	R	M	N.A	N.A	N.A	N.A	1, 2, 3, 4
~~b. Phase "B" Isolation~~								
~~1) Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2, 3, 4~~
~~2) Automatic Actuation Logic Actuation Relays~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2, 3, 4~~
~~3) Containment Pressure-High-3~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
b. ~~c.~~ Containment Ventilation ~~Purge and Exhaust~~ Isolation								
~~1) Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
1) ~~2)~~ Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4, 6
2) ~~3)~~ Safety Injection	See ~~Item~~ Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
3) Containment Area Radiation (Low Range) (RE-0002, RE-0003)	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3, 4, 6

Insert for page 3/4 3-42

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4) Containment Ventilation Radiation								
(i) Particulate Activity (RE-2565A)	S	R	M	N. A.	N. A.	N. A.	N. A.	1, 2, 3, 4, 6
(ii) Iodine Activity (RE-2565B)	S	R	M	N. A.	N. A.	N. A.	N. A.	1, 2, 3, 4, 6
(iii) Gaseous Activity (RE-2565C)	S	R	M	N. A.	N. A.	N. A.	N. A.	1, 2, 3, 4, 6

Loop 1	Loop 2	Loop 3	Loop 4
PI-0514	PI-[illegible]	PI-[illegible]	PI-[illegible]
PI-[illegible]	PI-[illegible]	PI-0535	PI-[illegible]
PI-[illegible]	PI-[illegible]	PI-[illegible]	PI-[illegible]

Loop 1	Loop 2	Loop 3	Loop 4
LT-0519	LT-0529	LT-[illegible]	LT-0549
LT-[illegible]	LT-[illegible]	LT-[illegible]	LT-0548
LT-[illegible]	LT-0527	LT-[illegible]	LT-0547
LT-0551	LT-0552	LT-[illegible]	LT-[illegible]

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2*(PI-0934, PI-0935, PI-0936)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Flow in Two Steam Lines-High Coincident With~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A~~	~~N.A.~~	~~N.A~~	~~1, 2, 3~~
~~T_{avg}-Low-Low~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2, 3~~
d ~~e~~. Steam Line Pressure-Low	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3
e ~~f~~. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip and Feedwater Isolation								
a. Safety Injection	See Functional Unit 1 above for all Safety Injection Surveillance Requirements.							
b ~~a~~. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
c ~~b~~. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
6. Auxiliary Feedwater								
~~a. Manual Initiation~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
a ~~b~~. Automatic Actuation and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3

*See also Specification [illegible]

Loop 1	Loop 2	Loop 3	Loop 4
LT-0519	LT-0529	LT-0539	LT-0549
LT-0518	LT-0528	LT-0538	LT-0548
LT-0517	LT-0527	LT-0537	LT-0547
LT-0551	LT-0552	LT-0553	LT-0554

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
b.c. Steam Generator Water Level-Low-Low*	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
~~d. Undervoltage - RCP~~	~~N.A.~~	~~R~~	~~N.A.~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A~~	~~1, 2~~
c e. Safety Injection	See ~~Item~~ Functional Unit 1. above for all Safety Injection Surveillance Requirements.							
d f. ~~Loss-of-Offsite Power~~ or Degraded 4.16 kV ESF Bus Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A	1, 2, 3
e g. Trip of ~~All~~ Main Feed water Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
~~h. Suction Transfer on Low Pressure~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
7. Automatic Switchover to Containment Emergency Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level-Low-Low* (LI-990, LI-991, LI-992, LI-993) Coincident With	S	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3, 4
~~Containment Sump Level-High~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3, 4~~
and Safety Injection	See ~~Item~~ Functional Unit 1. above for all Safety Injection Surveillance Requirements.							

* See also Specification 4.3.3.6.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power to 4.16 kV ESF Bus								
a. 4.16 kV ESF Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV ESF Bus Undervoltage (Grid Degraded Voltage)	N.A.	R.	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11 (PI-0455, PI-0456, PI-0457)	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~b. Low Low T_{avg}, P-12~~	~~N.A.~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
b. ~~c.~~ Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
~~d. Steam Generator Water Level, P-14~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~M(1)~~	~~M(1)~~	~~Q~~	~~1, 2, 3~~

See insert for 3/4 3-45.

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

VOGTLE-UNIT 1

INSERT TO PAGE 3/4 3-45 (TABLE 4.3-2)

10. CONTROL ROOM VENTILATION EMERGENCY MODE ACTUATION								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All Modes
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	All Modes
c. Safety Injection	See FUNCTIONAL UNIT 1 above for all Safety Injection Surveillance Requirements							
d. Intake Radiogas Monitor (RE-12116, RE-12117)	S	R	M	N.A.	N.A.	N.A.	N.A.	All Modes

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels for plant operations shown in Table 3.3-6 shall be OPERABLE with their Alarm/Trip Setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel for plant operations shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Atmosphere Radioactivity-High	1	2	All	≤ [2] mR/h	26
b. RCS Leakage Detection					
1) Particulate Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2) Gaseous Radioactivity	N.A.	1	1, 2, 3, 4	N.A.	29
2. Purge and Exhaust Ventilation					
a. Particulate Radioactivity	1	2	All	*	26
b. Gaseous Radioactivity	1	2	All	*	26
3. Fuel Storage Pool Areas					
a. Radioactivity-High					
Gaseous Radioactivity	1	2	**	≤ [2] mR/h	27
b. Criticality-Radiation Level	1	2	***	≤ 15 mR/h	28
4. Control Room					
a. Air Intake-Radiation Level	1/intake	2/intake	All	≤ [2] mR/h	27
b. Control Room Atmosphere Radiation-High	1	2	All	≤ [2] mR/h	27

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
5. Process Monitors					
Noble Gas Effluent Monitors					
i. Radwaste Building Exhaust System	1	1, 2, 3 & 4	( ) rad/hr	1-10^2 uCi/cc	30
ii. Auxiliary Building Exhaust System	1	1, 2, 3 & 4	( ) rad/hr	1-10^3 uCi/cc	30
iii. Steam Safety Valve Discharge	1/valve	1, 2, 3 & 4	( ) rad/hr	1-10^3 uCi/cc	30
iv. Atmospheric Steam Dump Valve Discharge	1/valve	1, 2, 3 & 4	( ) rad/hr	1-10^3 uCi/cc	30
v. Shield Building Exhaust System	1	1, 2, 3 & 4	( ) rad/hr	1-10^4 uCi/cc	30
vi. Containment Purge & Exhaust System	1	1, 2, 3 & 4	( ) rad/hr	1-10^5 uCi/cc	30
vii. Condenser Exhaust System	1	1, 2, 3 & 4	( ) rad/hr	1-10^5 uCi/cc	30

*Alarm/Trip Setpoint of 10 rad/hr is acceptable.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION
1. Containment					
a. Containment Area (Low Range) RE-0002, RE-0003	1	2	1, 2, 3, 4, 6*	See Table 3.3-4	See Table 3.3-3
b. Containment Area (High Range) RE-0005, RE-0006	1	2	1, 2, 3, 4	See Table 3.3-4	See Table 3.3-3
c. RCS Leakage Detection					
1) Gaseous Activity RE-2562C	1	1	1, 2, 3, 4	≤2 x background	26
2) Particulate Activity RE-2562A	1	1	1, 2, 3, 4	≤2 x background	26
2. Containment Ventilation					
a. Gaseous Activity RE-2565C	1	1	1, 2, 3, 4, 6*	See Table 3.3-4	See Table 3.3-3
b. Particulate Activity RE-2565A	1	1	1, 2, 3, 4, 6*	See Table 3.3-4	See Table 3.3-3
c. Iodine Activity RE-2565B	1	1	1, 2, 3, 4. 6*	See Table 3.3-4	See Table 3.3-3
3. Control Room Air Intake RE-12116, RE-12117	1	2	1, 2, 3, 4, 5** , 6**	See Table 3.3-4	See Table 3.3-3
4. Volume Reduction Room Selective Cubicle Monitors					
a. Gaseous Activity ARE-13133C	1	1	#	(Later)	27
b. Particulate Activity ARE-13133A	1	1	#	(Later)	27
c. Iodine Activity ARE-13133B	1	1	#	(Later)	27

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* During movement of irradiated fuel or movement of loads over irradiated fuel within containment.

** During movement of irradiated fuel or movement of loads over irradiated fuel.

\# During operation of the Volume Reduction System.

ACTION STATEMENTS

ACTION 26 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, satisfy the ACTION requirement of Specification 3.4.6.1.

ACTION 27 - With the number of OPERABLE channels less than the Minimum Channels OPERABLE requirement, suspend operation of the Volume Reduction System.

TABLE 3.3-6 (Continued)

TABLE NOTATIONS

* Must satisfy Specification 3.11.2.1 requirements.

** With irradiated fuel in the fuel storage pool areas.

*** With fuel in the fuel storage pool areas.

ACTION STATEMENTS

ACTION 26 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 27 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode.

ACTION 28 - With less than the Minimum Channels OPERABLE requirement, operation may continue for up to 30 days provided an appropriate portable continuous monitor with the same Alarm Setpoint is provided in the fuel storage pool area. Restore the inoperable monitors to OPERABLE status within 30 days or suspend all operations involving fuel movement in the fuel storage pool areas.

ACTION 29 - Must satisfy the ACTION requirement for Specification 3.4.6.1.

ACTION 30 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to OPERABLE status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Atmosphere Radioactivity-High	S	R	M	All
b. RCS Leakage Detection				
1) Particulate Radio-activity	S	R	M	1, 2, 3, 4
2) Gaseous Radioactivity	S	R	M	1, 2, 3, 4
2. Purge and Exhaust Ventilation				
a. Particulate Radioactivity	S	R	M	All
b. Gaseous Radioactivity	S	R	M	All
3. Fuel Storage Pool Areas				
a. Radioactivity-High-Gaseous Radioactivity	S	R	M	**
b. Criticality-Radiation Level	S	R	M	*
4. Control Room				
a. Air Intake Radiation Level	S	R	M	All
b. Control Room Atmosphere Radiation-High	S	R	M	All

TABLE NOTATIONS

* With fuel in the fuel storage pool area.
** With irradiated fuel in the fuel storage pool areas.

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Process Monitors				
Noble Gas Effluent Monitors				
i. Radwaste Building Exhaust System	S	R	M	1, 2, 3 & 4
ii. Auxiliary Building Exhaust System	S	R	M	1, 2, 3 & 4
iii. Steam Safety Valve Discharge	S	R	M	1, 2, 3 & 4
iv. Atmospheric Steam Dump Valve Discharge	S	R	M	1, 2, 3 & 4
v. Shield Building Exhaust System	S	R	M	1, 2, 3 & 4
vi. Containment Purge & Exhaust System	S	R	M	1, 2, 3 & 4
vii. Condenser Exhaust System	S	R	M	1, 2, 3 & 4

*Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Containment				
a. Containment Area (Low Range) RE-0002, RE-0003	See Table 4.3-2			
b. Containment Area (High Range) RE-0005, RE-0006	See Table 4.3-2			
c. RCS Leakage Detection				
1) Gaseous Activity RE-2562C	S	R	M	1, 2, 3, 4
2) Particulate Activity RE-2562A	S	R	M	1, 2, 3, 4
2. Containment Ventilation				
a. Gaseous Activity RE-2565C	See Table 4.3-2			
b. Particulate Activity RE-2565A	See Table 4.3-2			
c. Iodine Activity RE-2565B	See Table 4.3-2			
3. Control Room Air Intake RE-12116, RE-12117	See Table 4.3-2			
4. Volume Reduction Room Selective Cubicle Monitors				
a. Gaseous Activity ARE-13133C	S	R	M	*
b. Particulate Activity ARE-13133A	S	R	M	*
c. Iodine Activity ARE-13133B	S	R	M	*

* During operation of the Radwaste Solidification System

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

a. At least 75% of the detector thimbles,

b. A minimum of two detector thimbles per core quadrant, and

c. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $\cancel{FQ(Z)}$ $F_Q(Z)$ and F_{xy}.

ACTION:

With the Movable Incore Detection System inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The Movable Incore Detection System shall be demonstrated OPERABLE ~~at least once per 24 hours by normalizing each detector output when required for~~:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{xy}.

within 24 hours prior to use by irradiating each detector used and determining the acceptability of its voltage curve when required for:

INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.3 The seismic monitoring instrumentation shown in Table 3.3-7* shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required seismic monitoring instruments inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the instrument(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.3.1 Each of the above required seismic monitoring instruments shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-4.

which is accessible during power operations and which is

4.3.3.3.2 Each of the above required seismic monitoring instruments actuated during a seismic event greater than or equal to [0.01] g shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 ~~10~~ days following the seismic event. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 14 days describing the magnitude, frequency spectrum, and resultant effect upon facility features important to safety.

See insert to 3/4 3-51.

[*The instrumentation may be shared with additional units at a common ~~command~~ site provided seismic instrumentation and corresponding Technical Specifications meet the recommendations of Regulatory Guide 1.12, Revision 1, April 1974.]

Insert to 3/4 3-51

Each of the above seismic monitoring instruments which is actuated during a seismic event greater than or equal to 0.01g but is not accessible during power operation shall be restored to OPERABLE status and a CHANNEL CALIBRATION performed the next time the plant enters MODE 3 or below. A supplemental report shall then be prepared and submitted to the Commission within 10 days pursuant to Specification 6.9.2 describing the additional data from these instruments.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE	INSTRUMENT TAG NUMBER
1. Triaxial Time-History Accelerographs			
a. Free Field (500 ft from containment)	-1 to 1g	1	AXT-19900
b. Unit 1 Containment Gallery (basemat)	-1 to 1g	1	AXT-19901
c. Unit 1 Containment Operating Floor	-1 to 1g	1	AXT-19902
d. Auxiliary Building Basemat	-1 to 1g	1	AXT-19906
e. ~~d.~~ Unit 1 Containment Pressurizer Support	-1 to 1g	1	AXT-19903
2. Triaxial Peak Accelerographs			
a. Reactor Coolant Pump Motor (210 ft)	0 to 2g	1	AXR-19910
b. Steam Generator (185 ft)	0 to 2g	1	AXR-19911
c. NSCW Piping Outside Aux Bldg (220 ft)	0 to 2g	1	AXR-19913
d.		~~1~~	
e.		~~1~~	
3. Triaxial Seismic Switches			
a. Containment Tendon Gallery (basemat)	0.036 to 0.36g	1*	AXSH-19920
b.		~~1*~~	
c.		~~1*~~	
d.		~~1*~~	
4. Triaxial Response-Spectrum ~~Recorders~~ Analyzer			
a. Control Room	1.0 to 9.99g	1*	AXA-19980
b.		~~1~~	
c.		~~1~~	
d.		~~1~~	
e.		~~1~~	
f.		~~1~~	

*With reactor control room indication

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	ACCESSIBLE DURING MODES
1. Triaxial Time-History Accelerographs				
a. Free Field (500 ft from containment)	M*	R	SA	All
b. Unit 1 Containment Gallery (basemat)	M*	R	SA	All
c. Unit Containment Operating Floor	M*	R	SA	All
d. Auxiliary Building Basemat	M*	R	SA	All
e. ~~d.~~ Unit 1 Containment Pressurizer Support	M*	R	SA	5, 6
2. Triaxial Peak Accelerographs				
a. Reactor Coolant Pump Motor (210 ft)	N.A.	R	N.A.	5, 6
b. Steam Generator (195 ft)	N.A.	R	N.A.	5, 6
c. NSCW Piping Outside Aux. Bldg. (220 ft)	N.A.	R	N.A.	All
d. ______	N.A.	R	N.A.	
e. ______	N.A.	R	N.A.	
3. Triaxial Seismic Switches				
a. Containment Tendon Gallery (basemat) **	M	R	SA	All
b. ______ **	M	R	SA	
c. ______ **	M	R	SA	
d. ______ **	M	R	SA	
4. Triaxial Response-Spectrum ~~Recorders~~ Analyzer				
a. Control Room **	M	R	N.A. ~~SA~~	All
b. ______	N.A.	R	SA	
c. ______	N.A.	R	SA	
d. ______	N.A.	R	SA	
e. ______	N.A.	R	SA	
f. ______	N.A.	R	SA	

~~*Except seismic trigger.~~

**With reactor control room indications.

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.4 The meteorological monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE.

APPLICABILITY: At all times.

ACTION:

a. With one or more required meteorological monitoring channels inoperable for more than 7 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.4 Each of the above meteorological monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-5.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. Wind Speed		
a. Lower, Primary Tower	Nominal Elev. 10 m	1
b. Upper, Primary Tower	Nominal Elev. 60 m	1
2. Wind Direction		
a. Lower, Primary Tower	Nominal Elev. 10 m	1
b. Upper, Primary Tower	Nominal Elev. 60 m	1
3. Air Temperature - ΔT		
a. ΔT, Primary Tower	Nominal Elev. 10 m-60 m	1
~~b. ________~~	~~Nominal Elev. ________~~	~~1~~

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Wind Speed		
a. Nominal Elev. 10 m	D	SA
b. Nominal Elev. 60 m	D	SA
2. Wind Direction		
a. Nominal Elev. 10 m	D	SA
b. Nominal Elev. 60 m	D	SA
3. Air Temperature - ΔT		
a. Nominal Elev. 10 m	D	SA
b. Nominal Elev. 60 m	D	SA

INSTRUMENTATION

REMOTE SHUTDOWN SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.5.a The Remote Shutdown System ~~transfer switches, power, controls and~~ monitoring instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown monitoring channels less than the Minimum Channels OPERABLE as required by Table 3.3-9, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With one or more Remote Shutdown System transfer switches, ~~power~~, or control circuits inoperable, restore the inoperable switch(s)/circuit(s) to OPERABLE status within 7 days, or be in HOT STANDBY within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 Each remote shutdown monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-6.

4.3.3.5.2 Each Remote Shutdown System transfer switch, ~~power~~ and control circuit shall be demonstrated OPERABLE at least once per 18 months by verifying its capability to perform its intended function(s).

as required by 3.3.3.5.b

3.3.3.5.b The Remote Shutdown System transfer switches and controls of system components required for 1) reactivity control, 2) RCS pressure and inventory control, and 3) decay heat removal.

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM

INSTRUMENT [Illustrational only]	READOUT LOCATION	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Power Range Neutron Flux		2	2
2. Intermediate Range Neutron Flux		2	2
3. Source Range Neutron Flux		2	2
4. Reactor Trip Breaker Indication		1/trip breaker	1/trip breaker
5. Reactor Coolant Temperature - Average		2	2
6. Reactor Coolant Flow Rate		2	2
7. Pressurizer Pressure		2	2
8. Pressurizer Level		2	2
9. Steam Generator Pressure		2	2
10. Steam Generator Water Level		2/stm. gen.	2/stm. gen.
11. Control Rod Position Limit Switches		2/stm. gen.	2/stm. gen.
		1/insertion limit switch/rod	1/insertion limit switch/rod
12. RHR Flow Rate		2	2
13. RHR Temperature		2	2
14. Auxiliary Feedwater Flow Rate		2	2

TRANSFER SWITCHES [Illustrational Only]	SWITCH LOCATION
1. Auxiliary Feedwater Control	
2. Safe Shutdown Equipment Power	
a. Auxiliary Feedwater	
b. Charging	
c. Pressurizer Heaters	
d. Valves	
3. CVCS Makeup Flow Control	
4. Diesel Generator Control	
5. Electrical Distribution System Control	

CONTROL CIRCUITS [Illustrational Only]	SWITCH LOCATION
1. Auxiliary Feedwater Flow	
2. Pressurizer Heaters	
3. CVCS Makeup Flow	
4. Diesel Generator	
5. Electrical Distribution System	

TABLE 3.3-9
REMOTE SHUTDOWN SYSTEM

	Instrument Function	Readout Location[1]	Channels Available	Minimum Channels Operable
1.	Source Range Neutron Flux	A	1 NI 31E	1
2.	Extended Range Neutron Flux	B	1 NI 13135 C&D	1
3.	RCS LP Cold Leg Temperature	A, B	1/Loop Loop 1 TI 0413D (Panel A) Loop 2 TI 0423D (Panel B) Loop 3 TI 0433D (Panel B) Loop 4 TI 0443D (Panel A)	1/Loop
4.	RCS Hot Leg Temperature	A	2 Loop 1 TI 0413L Loop 4 TI 0443C	2
5.	Core Exit Thermocouples	B	2 Loop 2 Core Quadrant TI 10055 Loop 3 Core Quadrant TI 10056	2
6.	RCS Wide Range Pressure	A, B	2 Panel A PI 405A Panel B PI 403A	2
7.	Steam Generator Level Wide Range	A, B	1/Loop Loop 1 LI 501B (Panel A) Loop 2 LI 502B (Panel B) Loop 3 LI 503B (Panel B) Loop 4 LI 504B (Panel A)	1/Loop
8.	Pressurizer Level	A, B	2 Panel A LI 459C Panel B LI 460C	2
9.	RWST Level	L	1 LI 0990C	1[3]
10.	BAST Level	L	1 PI-10115[2]	1[3]
11.	CST Level	L	2 Tank 1 LI 5100 Tank 2 LI 5115	2[3]

---

1. A - Remote Shutdown Panel PSDA
   B - Remote Shutdown Panel PSDB
   L - Local Indication
2. Graph will be provided to determine level from pressure reading.
3. Alternate local level indication may be established to fulfill the minimum channels OPERABLE.

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Neutron Flux	M	Q
2. Intermediate Range Neutron Flux	M	N.A.
3. Source Range Neutron Flux	M	N.A.
4. Reactor Trip Breaker Indication	M	N.A.
5. Reactor Coolant Temperature - Average	M	R
6. Reactor Coolant Flow Rate	M	R
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Pressure	M	R
10. Steam Generator Water Level	M	R
11. Control Rod Position Limit Switches	M	R
12. RHR Flow Rate	M	R
13. RHR Temperature	M	R
14. Auxiliary Feedwater Flow Rate	M	R

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Source Range Neutron Flux	M	R
2. Extended Range Neutron Flux	M	R
3. RCS LP Cold Leg Temp	M	R
4. RCS Hot Leg Temp	M	R
5. Core Exit Thermocouples	M	R
6. RCS Wide Range Pressure	M	R
7. Steam Generator Level Wide Range	M	R
8. Pressurizer Level	M	R
9. RWST Level	M	R
10. BAST Level	M	R
11. CST Level	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION: As shown in Table 3.3-10.

a. ~~With the number of OPERABLE accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.~~

b. ~~With the number of OPERABLE accident monitoring instrumentation channels except the unit vent-high range noble gas monitor and the steam relief-high range radiation monitor less than the Minimum Channels OPERABLE requirements of Table 3.3-10, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.~~

c. ~~With the number of OPERABLE channels for the unit vent-high range noble gas monitor, or the steam relief-high range radiation monitor or the containment atmosphere-high range radiation monitor, or the reactor coolant radiation level monitor less than required by the Minimum Channels OPERABLE requirements, initiate an alternate method of monitoring the appropriate parameter(s), within 72 hours, and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status.~~

d. ~~The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT (Illustrational Only)	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Water Level - Wide Range	2/steam generator	1/steam generator
7. Reactor Coolant System Subcooling Margin Monitor	2	1
8. PORV Position Indicator	2/valve	1/valve
9. PORV Block Valve Position Indicator	2/valve	1/valve
10. Safety Valve Position Indicator	2/valve	1/valve
11. Containment Water Level (Narrow Range)	2	1
12. Containment Water Level (Wide Range)	2	1

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT [Illustrational Only]	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
13. In Core Thermocouples	4/core quadrant	2/core quadrant
14. Unit Vent - High Range Noble Gas Monitor	N.A.	1
15. Steam Relief - High Range Radiation Monitor	N.A.	1/steam line
16. Containment Atmosphere - High Range Radiation Monitor	2	1
17. Reactor Vessel Water Level	2	1
18. Reactor Coolant Radiation Level Monitor	2	1
19. Neutron Flux	2	1
20. Containment Isolation Valve Position	2/valve	1/valve
21. Containment Hydrogen Concentration	2	1
22. Condensate Storage Tank Level	2	1

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1.	Reactor Coolant Pressure (Wide Range) (PT-408, 418, 428, 438)	3	1	28
2.	Reactor Coolant System T_{hot} (Wide Range) (TI-413A, 423A, 433A, 443A)	1/loop	1/loop	30
3.	Reactor Coolant System T_{cold} (Wide Range) (TI-413B, 423B, 433B, 443B)	1/loop	1/loop	30
4.	SG Water Level (Wide Range) (LI-501, 502, 503, 504)	1/SG	1/SG	30
5.	SG Water Level (Narrow Range) (LI-517, 518, 519, 527, 528, 529, 537, 538, 539, 547, 548, 549)	3/SG	1/SG	28
6.	Pressurizer Level (LI-459, 460, 461)	3	1	28
7.	Containment Pressure (PI-934, 935, 936, 937)	3	1	28
8.	Steamline Pressure (PI-514, 515, 516, 524, 525, 526, 534, 535, 536, 544, 545, 546)	3/stm. line	1/stm. line	28
9.	RWST Level (LI-990, 991, 992, 993)	3	1	28
10.	Containment Normal Sumps Level (Narrow Range) (LI-7777, 7789)	2	1	29
11.	Containment Water Level (Wide Range) (LI-764, 765)	2	1	29
12.	Condensate Storage Tank Level (LI-5101, 5111, 5104, 5116)	2/tank	1/tank	29
13.	Auxiliary Feedwater Flow (FI-5152, 15152, 5153, 15153, 5151, 15151, 5150, 15150)	2/feed line	1/feed line	29
14.	Containment Radiation Level (High Range) (RE-0005, 0006)	2	1	31
15.	Steamline Radiation Monitor (RE-13119, 13120, 13121, 13122)	1/stm. line	1/stm. line	32
16.	Core Exit Thermocouples (TE-01-TE-50)	4/quad/train	2/quad/train	28
17.	Reactor Coolant System Subcooling	2	1	29
18.	Neutron Flux (Extended Range) (RE-13135 A1/A2, B1/B2)	2	1	29
19.	RVLIS	2	1	32
20.	Containment Hydrogen Concentration (AI-12979, 12980)	2	1	29
21.	Containment Pressure (Extended Range) (PI-10442, 10443)	2	1	29

TABLE 3.3-10 (Continued)

ACTION STATEMENTS

ACTION 28 - a. With the number of OPERABLE channels one less than the Required Number of Channels requirements, restore the inoperable channel to OPERABLE status within 31 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels two less than the Required Number of Channels requirement, restore at least one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status with 48 hours or be in HOT SHUTDOWN within the next 12 hours.

d. The provisions of Specification 3.0.4 are not applicable.

ACTION 29 - a. With the number of OPERABLE channels one less than the Required Number of Channels requirements restore one inoperable channel to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE channels less than the Minimum Channels Operable requirements, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 30 - With the number of OPERABLE channels less than the Minimum Channels Operable requirement, restore at least one inoperable channel to OPERABLE status within 48 hours, or be in HOT SHUTDOWN within the next 12 hours. The provisions of Specification 3.0.4 are not applicable.

See Insert to Page 3/4 3-62.

Insert to Page 3/4 3-62

31. With the number of OPERABLE channels less than the minimum channels OPERABLE requirement, initiate the alternate method of monitoring the parameter within 72 hours and either restore the inoperable channel(s) to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2, within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.

32. With the number of OPERABLE channels less than the required number of channels or the minimum channels OPERABLE requirement, restore the inoperable channel(s) to OPERABLE status as per Action 29a or b as applicable if repair is feasible during plant operation. If repair is not feasible, prepare and submit a Special Report to the Commission, pursuant to Specification 6.9.2 within 14 days that provides actions taken, cause of the inoperability, and the plans and schedule for restoring the channels to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.*

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* Action Statement 32 applies to the first fuel cycle only. Action Statement 29 is applicable thereafter.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT [Illustrational Only]	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Water Level - Wide Range	M	R
7. Reactor Coolant System Subcooling Margin Monitor	M	R
8. PORV Position Indicator	M	R
9. PORV Block Valve Position Indicator	M	R
10. Safety Valve Position Indicator	M	R
11. Containment Water Level (Narrow Range)	M	R
12. Containment Water Level (Wide Range)	M	R

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT [Illustrational Only]	CHANNEL CHECK	CHANNEL CALIBRATION
13. In Core Thermocouples	M	R
14. Unit Vent - High Range Noble Gas Monitor	M	R
15. Steam Relief - High Range Radiation Monitor	M	R
16. Containment Atmosphere - High Range Radiation Monitor	M	R*
17. Reactor Vessel Water Level	M	R
18. Reactor Coolant Radiation Level Monitor	M	R
19. Neutron Flux	M	R
20. Containment Is lation Valve Position	M	R
21. Containment Hydrogen Concentration	M	R
22. Condensate Storage Tank Level	M	R

*CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/h and a one point calibration check of the detector below 10 R/h with an installed or portable gamma source.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	Instrument	Channel Check	Channel Calibration
1.	Reactor Coolant Pressure (Wide Range) (PT-408, 418, 428, 438)	M	R
2.	Reactor Coolant System T_{hot} (Wide Range) (TI-413A, 423A, 433A, 443A)	M	R
3.	Reactor Coolant System T_{cold} (Wide Range) (TI-413B, 423B, 433B, 443B)	M	R
4.	Steam Generator Water Level (Wide Range) (LI-501, 502, 503, 504)	M	R
5.	Steam Generator Water Level (Narrow Range) (LI-517, 518, 519, 527, 528, 529, 537, 538, 539, 547, 548, 549)	M	R
6.	Pressurizer Level (LI-459, 460, 461)	M	R
7.	Containment Pressure (PI-934, 935, 936, 937)	M	R
8.	Steamline Pressure (PI-514, 515, 516, 524, 525, 526, 534, 535, 536, 544, 545, 546)	M	R
9.	RWST Level (LI-990, 991, 992, 993)	M	R
10.	Containment Normal Sumps Level (Narrow Range) (LI-7777, 7789)	M	R
11.	Containment Water Level (Wide Range) (LI-764, 765)	M	R
12.	Condensate Storage Tank Level (LI-5101, 5111, 5104, 5116)	M	R
13.	Auxiliary Feedwater Flow (FI-5152, 15152, 5153, 15153, 5151, 15151, 5150, 15150)	M	R
14.	Containment Radiation Level (High Range) (RE-0005, RE-0006)	M	R
15.	Steamline Radiation Monitor (RE-13119, 13120, 13121, 13122)	M	R
16.	Core Exit Thermocouples (TE-01-TE-50)	M	R
17.	Reactor Coolant System Subcooling	M	R
18.	Neutron Flux (Extended Range) (RE-13135 A1/A2, B1/B2)	M	R
19.	RVLIS	M	R
20.	Containment Hydrogen Concentration (AI-12979, 12980)	M	R
21.	Containment Pressure (Extended Range) (PI-10442, 10443)	M	R

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent Chlorine Detection Systems, with their Alarm/Trip Setpoints adjusted to actuate at a chlorine concentration of less than or equal to ~~5~~ 2 ppm, shall be OPERABLE.

APPLICABILITY: ~~All MODES.~~ Modes 1, 2, 3, and 4. Modes 5 and 6 during movement of irradiated fuel or movement of loads over irradiated fuel.

ACTION:

a. With one Chlorine Detection System inoperable, restore the inoperable system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room Emergency Ventilation System in the ~~recirculation~~ isolation mode of operation.

b. With both Chlorine Detection Systems inoperable, within 1 hour initiate and maintain operation of the Control Room Emergency Ventilation System in the ~~recirculation~~ isolation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each Chlorine Detection System shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

4.3.3.8.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION [Illustrative]	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
1. Containment **			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
2. Control Room			
3. Cable Spreading			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
4. Computer Room			
5. Switchgear Room			
6. Remote Shutdown Panels			
7. Station Battery Rooms			
8. Turbine			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
9. Diesel Generator			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
10. Safety-Related Pumps			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			
11. Fuel Storage			
a. Zone 1 Elevation ___			
b. Zone 2 Elevation ___			

[List all detectors in areas required to ensure the OPERABILITY of safety-related equipment.]

---

*(x/y): x is number of Function A (early warning fire detection and notification only) instruments.
y is number of Function B (actuation of Fire Suppression Systems and early warning and notification) instruments.

**The fire detection instruments located within the containment are not required to be OPERABLE during the performance of Type A containment leakage rate tests.

INSTRUMENTATION

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.~~9~~8 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.~~9~~8 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and

c. A CHANNEL CALIBRATION at least once per 18 months.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.~~10~~ 9 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-~~12~~ 11 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specification 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-~~12~~ 11. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4, are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.~~10~~ 9 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-8.

TABLE 3.3-~~12~~ 11

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
a. Liquid Radwaste Effluent Line (RE-0018)	1	~~35~~ 33
b. Steam Generator Blowdown Effluent Line (RE-0021)	1	~~36~~ 34
c. Turbine Building (Floor Drains) Sumps Effluent Line (RE-0848)	1	~~36~~ 34
d. Control Building (Floor Drains) Sumps Effluent Line (RE-17646)		35
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release		
a. Nuclear Service Cooling Water System Effluent Line (RE-0020 A & B)	1	~~37~~ 35
~~b. Component Cooling Water System Effluent Line~~	~~1~~	~~37~~
~~3. Continuous Composite Samplers and Sampler Flow Monitor~~		
~~a. Steam Generator Blowdown Effluent Line (alternate to Item 1.b.)~~	~~1~~	~~36~~
~~b. Turbine Building Sumps Effluent Line (alternate to Item 1.c.)~~	~~1~~	~~36~~
4. Flow Rate Measurement Devices		
a. Liquid Radwaste Effluent Line (FI-0018)	1	~~38~~ 36
b. Steam Generator Blowdown Effluent Line (FI-0021)	1	~~38~~ 36
~~c. Discharge Canal~~ c. Flow to Blowdown Sump (FI-7620C)	1	~~38~~ 36

TABLE 3.3-~~12~~ 11 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
~~5. Radioactivity Recorders*~~		
~~a. Liquid Radwaste Effluent Line~~	~~1~~	~~35~~
~~b. Steam Generator Blowdown Effluent Line~~	~~1~~	~~39~~

~~*Required only if Alarm/Trip Setpoint is based on recorder-controller.~~

TABLE 3.3-~~12~~ 11 (Continued)

ACTION STATEMENTS

ACTION ~~35~~ 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION ~~36~~ 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 microCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 microCurie/gram DOSE EQUIVALENT I-131.

ACTION ~~37~~ 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower limit of detection of no more than 10^{-7} microCurie/ml.

ACTION ~~38~~ 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

~~ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluents releases via this pathway may continue provided the radioactivity level is determined at least once per 4 hours during actual releases.~~

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluent Line (RE-0018)	D	P	R(3)	Q(1)
b. Steam Generator Blowdown Effluent Line (RE-0021)	D	M	R(3)	Q(1)
c. Turbine Building (Floor Drains) Sumps Effluent Line (RE-0848)	D	M	R(3)	Q(1)
d. Control Building (Floor Drains) Sumps Effluent Line (RE-[illegible])	D	M	R(3)	Q(1)
2. Radioactivity Monitors Providing Alarm But Not Providing Automatic Termination of Release				
a. Nuclear Cooling Service Water System Effluent Line (RE-0020 A+B)	D	M	R(3)	Q(2)
~~b. Component Cooling Water System Effluent Line~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~
~~3. Continuous Composite Samplers and Sampler Flow Monitor~~				
~~a. Steam Generator Blowdown Effluent Line (alternate to Item 1.b.)~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Turbine Building Sumps Effluent Line (alternate to Item 1.c.)~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~

TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
4. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line (FI-0018)	D(4)	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line (FI-0021)	D(4)	N.A.	R	Q
~~c. Discharge Canal~~ c. Flow to Blowdown Sump (FI-7620)	D(4)	N.A.	R	Q
~~5. Radioactivity Recorders*~~				
~~a. Liquid Radwaste Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~

*Required only if Alarm/Trip Setpoint in based on recorder-controller.

TABLE 4.3-8 (Continued)

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occur if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.~~11~~ 10 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-~~13~~ 12 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.11.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.3-~~13~~ 12

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-~~13~~ 12. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.~~11~~ 10 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-9.

TABLE 3.3-~~13~~ 12

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. ~~WASTE GAS HOLDUP SYSTEM~~ GASEOUS WASTE PROCESSING SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ARE-0014)	1	***	~~45~~ 37
~~b. Iodine Sampler~~	~~1~~	~~*~~	~~51~~
~~c. Particulate Sampler~~	~~1~~	~~*~~	~~51~~
d. Effluent System Flow Rate Measuring Device (AFE-0014)	1	***	~~46~~ 38
~~e. Sampler Flow Rate Measuring Device~~	~~1~~	~~*~~	~~46~~
2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)			
a. Hydrogen Monitor (Automatic Control)	1	**	49
b. Hydrogen or Oxygen Monitor (Process)	1	**	49

TABLE 3.3-~~13~~ 12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
~~2B.~~ 1. GASEOUS WASTE PROCESSING ~~WASTE GAS HOLDUP~~ SYSTEM Explosive Gas Monitoring System (~~for systems not designed to withstand the effects of a hydrogen explosion~~)			
a. Hydrogen Monitor~~s~~ (Automatic control, redundant)	~~2~~ 1/recombiner	**	~~50, 52~~ 42
b. Hydrogen ~~or~~ Oxygen Monitor~~s~~ (Process, ~~dual~~)	~~2~~ 2/recombiner	**	~~50~~ 40
3. Air Ejector and Steam Packing Exhauster ~~Condenser Evacuation~~ System			
a. Noble Gas Activity Monitor (RE-12839C)	1	***	~~47~~ 59
b. Iodine Sampler (RE-12839B)	1	***	~~51~~ 41
c. Particulate Sampler (RE-12839A)	1	***	~~51~~ 41
d. Flow Rate Monitor (FI-12839)	1	***	~~46~~ 38
e. Sampler Flow Rate Monitor (FI-12839)	1	***	~~46~~ 38

TABLE 3.3-~~13~~ 12 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4.	~~Vent Header System~~ Plant Vent			
	a. Noble Gas Activity Monitor (RE-12442C or RE-12444C)	1	*	~~47~~ 39
	b. Iodine Sampler (RE-12442B or RE-12444B)	1	*	~~51~~ 41
	c. Particulate Sampler (RE-12442A or RE-12444A)	1	*	~~51~~ 41
	d. Flow Rate Monitor (FI-12442)	1	*	~~46~~ 38
	e. Sampler Flow Rate Monitor (FI-12442 or FI-12444)	1	*	~~46~~ 38
~~5.~~	~~Containment Purge System~~			
	~~a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release~~	~~1~~	~~*~~	~~48~~
	~~b. Iodine Sampler~~	~~1~~	~~*~~	~~51~~
	~~c. Particulate Sampler~~	~~1~~	~~*~~	~~51~~
	~~d. Flow Rate Monitor~~	~~1~~	~~*~~	~~46~~
	~~e. Sampler Flow Rate Monitor~~	~~1~~	~~*~~	~~46~~

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
6. Auxiliary Building Ventilation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46
7. Fuel Storage Area Ventilation System			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	51
c. Particulate Sampler	1	*	51
d. Flow Rate Monitor	1	*	46
e. Sampler Flow Rate Monitor	1	*	46

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TABLE 3.3-~~13~~ (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
Solidification Building 8. Radwaste Area Ventilation System			
a. Noble Gas Activity Monitor (ARE-0026C)	1	*	~~47~~ 39
b. Iodine Sampler (ARE-0026B)	1	*	~~51~~ 41
c. Particulate Sampler (ARE-0026A)	1	*	~~51~~ 41
d. Flow Rate Monitor (AFI-0026)	1	*	~~46~~ 38
e. Sampler Flow Rate Monitor (AFI-0026)	1	*	~~46~~ 38
~~9. Other Exhaust and Vent Systems such as:~~			
~~Steam Generator Blowdown Vent System and Turbine Gland Seal Condenser Exhaust~~			
~~a. Noble Gas Activity Monitor~~	~~1~~	~~*~~	~~47~~
~~b. Iodine Sampler~~	~~1~~	~~*~~	~~51~~
~~c. Particulate Sampler~~	~~1~~	~~*~~	~~51~~
~~d. Flow Rate Monitor~~	~~1~~	~~*~~	~~46~~
~~e. Sampler Flow Rate Monitor~~	~~1~~	~~*~~	~~46~~

TABLE 3.3-~~13~~ 12 (Continued)

TABLE NOTATIONS

* At all times.

** During ~~WASTE GAS HOLDUP~~ GASEOUS WASTE PROCESSING SYSTEM operation.

*** During radioactive releases via this pathway.

ACTION STATEMENTS

ACTION ~~45~~ 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION ~~46~~ 38 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION ~~47~~ 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

~~ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.~~

~~ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this WASTE GAS HOLDUP SYSTEM may continue provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.~~

~~ACTION 50 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.~~

See insert for page 3/4 3-78.

Insert for page 3/4 3-73

ACTION 40 - a. With the outlet oxygen monitor channel inoperable, operation of the system may continue provided grab samples are taken and analyzed at least once per 24 hours and the oxygen concentration remains less than 1 percent.

b. With the inlet oxygen monitor inoperable, operation may continue if inlet hydrogen monitor is OPERABLE.

c. With both oxygen channels or both of the inlet oxygen and inlet hydrogen monitors inoperable, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

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TABLE 3.3-~~13~~ (Continued)

TABLE NOTATIONS (Continued)

ACTION ~~51~~ 41 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

~~ACTION 52 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner.~~

See insert to page 3/4 3-79.

Insert for page 3/4 3-79

ACTION 42 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, suspend oxygen supply to the recombiner. Addition of waste gas to the system may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations or at least once per 24 hours during other operations and the oxygen concentration remains less than 1 percent.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. ~~WASTE GAS HOLDUP SYSTEM~~ GASEOUS WASTE PROCESSING					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (ARE-0014)	P	P	R(3)	Q(1)	***
b. ~~Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
c. ~~Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
b. ~~d.~~ Effluent System Flow Rate Measuring Device (AFI-0014)	P	N.A.	R	Q	***
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~2A. WASTE GAS HOLDUP SYSTEM Explosive Gas Monitoring System (for systems designed to withstand the effects of a hydrogen explosion)~~					
~~a. Hydrogen Monitor (Automatic Control)~~	~~D~~	~~N.A.~~	~~Q(4)~~	~~M~~	~~**~~
~~b. Hydrogen or Oxygen Monitor (Process)~~	~~D~~	~~N.A.~~	~~Q(4) or Q(5)~~	~~M~~	~~**~~

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
2~~B~~. GASEOUS WASTE PROCESSING ~~WASTE GAS HOLDUP~~ SYSTEM Explosive Gas Monitoring System ~~(for systems not designed to withstand the effects of a hydrogen explosion)~~					
a. Hydrogen Monitors ~~(Automatic Control, redundant)~~	D	N.A.	Q(4)	M	**
b. ~~Hydrogen or~~ Oxygen Monitors ~~(Process, dual)~~	D	N.A.	Q(4) ~~or Q(5)~~	M	**
3. Condenser ~~Evacuation~~ Air Ejector and Steam Packing Exhaust System					
a. Noble Gas Activity Monitor (RE-12839C)	D	M	R(3)	Q(2)	***
b. Iodine Sampler (RE-12839B)	W	N.A.	N.A.	N.A.	***
c. Particulate Sampler (RE-12839A)	W	N.A.	N.A.	N.A.	***
d. Flow Rate Monitor (FI-12839)	D(5)	N.A.	R	Q	***
e. Sampler Flow Rate Monitor (FI-12839)	D(5)	N.A.	R	Q	***
4. Plant Vent Header System					
a. Noble Gas Activity Monitor (RE-12442C or RE-12444C)	D	M	R(3)	Q(2)	*
b. Iodine Sampler (RE-12442B or RE-12444B)	W	N.A.	N.A.	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Plant Vent Header System (Continued)					
c. Particulate Sampler (RE-12442A or RE-12444A)	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor (FI-12442)	D(5)	N.A.	R	Q	*
e. Sampler Flow Rate Monitor (FI-12442 or FI-12444)	D(5)	N.A.	R	Q	*
~~5. Containment Purge System~~					
~~a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release~~	~~D~~	~~P~~	~~R(3)~~	~~Q(1)~~	~~*~~
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~d. Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~6. Auxiliary Building Ventilation System~~					
~~a. Noble Gas Activity Monitor~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~	~~*~~

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Building Ventilation System (Continued)					
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
7. Fuel Storage Area Ventilation System					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. ~~8.~~ Radwaste ~~Area~~ Solidification Building Ventilation System					
a. ~~Noble Gas~~ Radiogas Activity Monitor (ARE-3026C)	D	M	R(3)	Q(2)	*
b. Iodine Sampler (ARE-0026B)	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler (ARE-0026A)	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor (AFI-0026)	D(5)	N.A.	R	Q	*
e. Sampler Flow Rate Monitor (AFI-0026)	D(5)	N.A.	R	Q	*
~~9. Other Exhaust and Vent Systems such as:~~					
~~Steam Generator Blowdown Vent System, Turbine Gland Seal Condenser Exhaust~~					
~~a. Noble Gas Activity Monitor~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~	~~*~~
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~d. Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~

TABLE 4.3-9 (Continued)

TABLE NOTATIONS

* At all times. GASEOUS WASTE PROCESSING

** During ~~WASTE GAS HOLDUP~~ SYSTEM operation.

*** During Releases via this pathway.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint, or

b. Circuit failure, or

c. Instrument indicates a downscale failure, or

d. Instrument controls not set in operate mode.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples ~~containing a nominal:~~ in accordance with the manufacturers' recommendations.

~~a. One volume percent hydrogen, balance nitrogen, and~~

~~b. Four volume percent hydrogen, balance nitrogen.~~

~~(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:~~

~~a. One volume percent oxygen, balance nitrogen, and~~

~~b. Four volume percent oxygen, balance nitrogen.~~

5 CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days which continuous, periodic, or batch releases are made.

INSTRUMENTATION

HIGH ENERGY LINE BREAK ISOLATION SENSORS

LIMITING CONDITION FOR OPERATION

3.3.3.~~12~~ 11 The high energy line break instrumentation listed in Table 3.3-14 shall be OPERABLE.

APPLICABILITY: As noted in Table 3.3-~~14~~ 13

ACTION:

a. With the number of OPERABLE electric steam boiler isolation instruments less than required by Table 3.3-~~14~~ 13, restore the inoperable instruments to OPERABLE status within 7 days or suspend operation of the electric steam boiler until the inoperable sensors are restored to OPERABLE status. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE steam generator blowdown line isolation instruments or letdown line isolation instruments less than required by Table 3.3-~~14~~ 13, restore the inoperable instruments to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.~~12~~ 11 Each of the above high energy line break isolation instruments shall be demonstrated OPERABLE by the performance of an ANALOG CHANNEL OPERATIONAL TEST at least once per 18 months.

TABLE 3.3-13

HIGH-ENERGY LINE BREAK INSTRUMENTATION

	Isolation Function	Instrument Channel	Minimum Channels Operable	Applicable Modes
1.	Electric Steam Boiler Isolation	TE 19722A (RD52) TE 19723A (RD52)	1	*
		TE 19722B (RC41) TE 19723B (RC41)	1	*
		TE 19722C (RC64) TE 19723C (RC64)	1	*
		TE 19722D (RC66) TE 19723D (RC66)	1	*
		FT 19722 FT 19723	1	*
		TE 19722E (RC95) TE 19723E (RC95)	1	*
2.	Steam Generator Blowdown Line Isolation	TE 15212A (RB08) TE 15216A (RB08)	1	1, 2, 3, 4
		TE 15212B (RC106) TE 15216B (RC106)	1	1, 2, 3, 4
		TE 15212C (RC107) TE 15216C (RC107)	1	1, 2, 3, 4
		TE 15212D (RC108) TE 15216D (RC108)	1	1, 2, 3, 4
		FT 15212A (Loop 1) FT 15216A	1	1, 2, 3, 4
		FT 15212B (Loop 2) FT 15216B	1	1, 2, 3, 4
		FT 15212C (Loop 3) FT 15216C	1	1, 2, 3, 4
		FT 15212P (Loop 4) FT 15216P	1	1, 2, 3, 4
3.	Letdown Line Isolation	TE 15214A (A07) TE 15215A (A07)	1	1, 2, 3, 4
		TE 15214B (A08) TE 15215B (A08)	1	1, 2, 3, 4
		TE 15214C (A09) TE 15215C (A09)	1	1, 2, 3, 4
		PT 15214 PT 15215	1	1, 2, 3, 4

* Required during all modes when electric steam boiler is in operation.

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2*, and 3*.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines, or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be maintained, calibrated, tested, and inspected in accordance with the VEGP Turbine Overspeed Protection Reliability Program. Adherence to this program shall demonstrate OPERABILITY of this system. The program and any revisions should be reviewed and approved in accordance with Specification 6.5.1.6. Revisions shall be made in accordance with the provisions of 10 CFR 50.59.

*Not applicable in MODE 2 or 3 with all main steam line isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam line(s) or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:

a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:

1) [Four] high pressure turbine stop valves,

2) [Four] high pressure turbine governor valves,

3) [Four] low pressure turbine reheat stop valves, and

4) [Four] low pressure turbine reheat intercept valves.

b. At least once per 31 days by direct observation of the movement of each of the above valves through one complete cycle from the running position,

c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems, and

d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks, and stems and verifying no unacceptable flaws or excessive corrosion. If unacceptable flaws or excessive corrosion are found, all other valves of that type shall be inspected.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.3

3/4.3.1:

A description of the reactor trip system is provided in section 7.2 of the VEGP FSAR.

Table 3.3-1:

General:

Notations such as *, # were changed to letters for clarification and to avoid confusion.

All references to three-loop and four-loop plants were deleted to reflect the plant-specific design of four loops.

All reference to three- and four-loop operation were deleted since the plant will only be operated in four-loop.

Instrument numbers were added to this table to improve the operator's understanding of Technical Specification compliance.

Cross references were added for those instruments which perform an ESFAS or PAMS function in addition to an RPS function. This was done as an aid for the operator in assuring compliance with the Technical Specifications.

Action statements were renumbered to be sequential following deletion of action statements not applicable to VEGP.

Specific:

Functional Unit 5 & 6: The minimum channels operable requirement for the source range and intermediate range instrumentation was reduced from 2 to 1 channel during startup on the following basis:

1. No credit was taken for operation of the trips associated with either the intermediate or source range instrumentation in the accident analyses.

2. The plant is in the startup mode for a limited period of time which further reduces the exposure to complete failure of the source range, intermediate range, and power range low setpoint trips in the event of an uncontrolled RCCA withdrawal.

Functional Units 7 & 8: The ** note was deleted since N-1 loop operation will not be permitted.

Functional Unit 14: The number of channels for undervoltage and underfrequency trip functions was modified to reflect the plant-specific design.

Functional Unit 14 (STS): The VEGP design does not include the steam feedflow mismatch with low steam generator level. This item was deleted.

Table Notations e and f were added to make the Technical Specifications consistent with the plant-specific analysis. Note f was applied to the Applicable Modes for Functional Units 9, 11, 14, and 15 and Note e was applied to the Applicable Modes for Functional Unit 16.

Action Statement 5: This action statement was revised to provide appropriate action when two channels are inoperable rather than applying the action required by section 3.0.3.

Action statement 7 (STS) was deleted per WCAP-10271/Supplement 1 which provides justification for testing one channel of a 2/3 logic with another channel failed provided one channel is in trip. Action 7 was applicable to 2/3 logic. By WCAP-10271/Supplement 1 Action 6 is used instead.

Action Statement 7: The word "annunciators" was deleted because the permissive lights are on status panels instead of annunciator panels.

Action Statement 9 (STS) is applicable to N-1 loop operation only. Vogtle will not operate in the N-1 configuration. Therefore, Action Statement 9 of NUREG-0452 was deleted.

Footnote g was added to the Table Notations and applied to Functional Unit 13, and Action statements 2, 6, and 10 were revised based on WCAP 10271 and supplement elements to that report. Maintenance outage times have been determined in accordance with WCAP 10271/Supplement 1 and are based upon maintaining an appropriate level of reliability of the reactor protection system and engineered safety features instrumentation.

Table 3.3-2:

Instrument numbers were added to this table to improve the operator's understanding of Technical Specification compliance.

Steam/feedwater flow mismatch with low steam generator level was deleted to reflect the absence of this function in the VEGP plant-specific design.

Table 4.3-1:

General:

Notations such as *, # were changed to letters for clarification and to avoid confusion.

Analog Channel Operation Test Column and Trip Actuating Device Operational Test Column: Specified surveillance intervals have been determined in accordance with WCAP-10271/Supplement 1 and supplements to that report and are based upon maintaining an appropriate level of reliability of the reactor protection system and engineered safety features instrumentation. Table Notations 12, 8, and 13 were revised/added based on WCAP 10271/Supplement 1. STS Table Notation 8 was deleted since it is no longer required as noted in the SER for WCAP 10271/Supplement 1.

Instrument numbers were added to this table to improve the operator understanding of Technical Specification compliance.

Specific:

Functional Unit 6: STS Table Notation 12 was deleted from this functional unit and from the Table Notations on the basis that the VEGP design does not include a flux doubling signal or any auto-actuation on flux doubling. See subsection 15.4.6 of the VEGP FSAR.

Functional Unit 9: Footnote e was added to Functional Unit 9 and to the Table Notations to clarify that this trip is blocked below the P-7 setpoint.

Functional Unit 11: A cross reference to Specification 4.3.3.6 was added since this instrumentation performs a PAMS function as well as an RPS function. This was done as an aid to the operator in assuring compliance with the Technical Specifications.

Functional Unit 14 (STS): Steam/feedwater flow mismatch with low steam generator level and low power reactor trip block were deleted to reflect the absence of these functions in the VEGP plant-specific design.

Functional Unit 16 was revised to reflect plant-specific design. The low fluid oil pressure is derived from an analog device (pressure transmitter). Footnote b was added to the Applicable Modes and to the Table Notations to reflect the plant-specific design and analysis.

Functional Unit 18: P-7 was deleted from this item in accordance with the SER on WCAP 10271.

Functional Unit 23 (STS) and Table Notations 14, 15, and 16: These items were revised/deleted per the recommendations of the WOG in response to Generic Letter 85-09. Westinghouse recommends eliminating the bypass breaker Technical Specification surveillances in Generic Letter 85-09 as presented at the WOG meeting in San Antonio.

Table Notations 3 and 6 were modified to change the surveillance from a calendar-based frequency to an EFPD frequency in recognition of the fuel burnup-related nature of this specification. That is, flux profiles change with burnup, not calendar time. Table Notation 6 has also been expanded to provide additional information concerning the nature of the incore-excore calibration.

Table Notation 5 was revised on the basis that the plateau curves should be evaluated against the most recent verifiable data. Initial plateau curves shall be measured for each detector. Subsequent plateau curves shall be obtained, evaluated, and compared to the initial curves.

Table Notation 8 has also been revised to be plant specific with regard to the source range high flux at shutdown alarm. See subsection 15.4.6 of the VEGP FSAR.

3/4.3.2:

A description of the engineered safety features actuation system is provided in section 7.3 of the VEGP FSAR and the logic diagrams are provided in section 7.2.

Table 3.3-3:

General:

Instrument numbers were added to these tables to improve the operator's understanding of Technical Specification compliance.

Notations such as * and # were changed to letters for clarification and to avoid confusion.

Cross references to Specification 3.3.3.6 were added for those instruments which also perform a PAMs function.

Action statements were renumbered to be consecutive following revision of the actions to make them plant specific.

Reference to three- and four-loop plants and three- and four-loop operation were deleted to reflect the plant-specific design of four loops and the fact that N-1 operation will not be permitted.

Specific:

Functional Unit 1: The description of this functional unit was revised to be plant specific.

STS Action Statement 15 was deleted per WCAP 10271 which provides justification for testing one channel of a 2/3 logic with another failed channel provided one channel is in trip. Action 15 was applicable to 2/3 logic. By WCAP 10271, the action statement which is numbered 18 in this draft is used instead.

"Differential Pressure Between Steam Lines-High" was deleted since this actuation signal is not included in the VEGP design.

The total No. of channels, channels to trip, and minimum channels operable requirements for "Steam Line Pressure-Low" were revised to be plant specific.

Functional Unit 3: "Containment Area Radiation (High Range)" was added as an initiating signal for "Phase "A" Isolation" reflecting the VEGP design.

"Phase "B" Isolation" was deleted since this isolation performs no function at VEGP. That is, no equipment is actuated on a Phase "B" Isolation.

"Purge and Exhaust Isolation" was revised to "Containment Ventilation Isolation" to reflect the VEGP design. The isolation of the purge system is accomplished via the containment ventilation isolation function. Mode 6 was added to the Applicable Modes to give recognition to the ventilation isolation function in mitigating the consequences of a fuel handling accident inside containment. Action Statement 16 was added for the Mode 6 case to be consistent with Specification 3.9.9 and Table Notation C was added and applied to Mode 6 to recognize the fact that if a fuel handling accident cannot occur due to other precautions, then this function's operability is not required in Mode 6. Containment ventilation isolation also occurs off of containment area radiation (low range) and containment ventilation radiation (particulate, iodine, and gaseous). Therefore, these instruments were added to be consistent with plant design.

Footnote g was added to the Table Notations in order to specify the minimum channels operable requirements for containment ventilation radiation. Any one of these three channels will provide a containment ventilation isolation. Therefore, the minimum channels operable requirements of at least two instruments will provide the necessary safety function and meet single-failure criteria. This will allow flexibility for maintenance without affecting the reliability of the containment ventilation isolation function.

Footnote d was applied to Action Statement 15 for the reasons discussed under 3/4.6.3.

Finally, manual initiation was deleted for containment ventilation isolation since a system level handswitch is not provided in the VEGP design.

Functional Unit 4: Footnote f was added to the Applicable Modes (Modes 2 and 3) and the Table Notations to give recognition to the lack of need for steam line isolation capability if the steam lines are already isolated.

Table Notations ** and *** were deleted on the basis that P-12 does not perform an ESF-related function and N-1 loop operation will not be permitted. In addition, since there are two MSIVs per steam line, manual initiation at the individual level was revised to reflect the fact that there are two channels per steamline - one for each MSIV.

Action Statement 15 (STS) was replaced with Action Statement 18 for the reasons discussed under Functional Unit 1.

"Steam Flow in Two Steam Lines-High" and "Steam Flow in Two Steam Lines-High Coincident With T -Low-Low" were deleted since these actuation signals are not included in the VEGP design.

The requirements for total No. of channels, channels to trip, and minimum channels operable for "Steam Line Pressure-Low" and "Steam Line Pressure-Negative Rate-High" were revised to be plant specific. Table Notation a was applied to Mode 3 for "Steam Line Pressure-Low" since plant design includes the feature that this function may be blocked below the P-11 setpoint.

Functional Unit 5: Safety injection was added as an actuation signal reflecting the VEGP specific design.

Function Unit 6: "Manual Initiation" was deleted on the basis that there is no specific handswitch which will initiate auxiliary feedwater manually. Manual initiation requires manually starting each pump, etc. See subsection 7.3.7 of the VEGP FSAR.

Initiation of auxiliary feedwater on a loss-of-offsite power is derived from four solid-state undervoltage devices in each train-related sequencer. The logic provided is two-out-of-four. Thus, a two-out-of-four undervoltage signal from a 4.16 kV emergency bus will result in a start signal to the associated motor-driven auxiliary feedwater pump and the turbine-driven pump. The turbine-driven pump gets signals from both trains, whereas a start signal from either train will start the turbine-driven pump. See paragraph 8.3.1.1.3 of the VEGP FSAR.

Concerning auxiliary feedwater initiation on trip of all main feedwater pumps, each main feedwater pump is equipped with a single pressure switch to sense pump trip. A two-out-of-two logic is provided. See FSAR figure 7.3.7-1 for details. Action Statement 25 was applied to this initiating signal based on the fact that no credit is taken for the start of the motor-driven auxiliary feedwater pumps on the loss of both main feedwater pumps in the accident analysis. This anticipatory function is bound by the accident analysis which depends on the steam generator low-low level trip to start the auxiliary feedwater pumps. Therefore, failure of the subject trip would not compromise the plant's protection system.

Suction transfer on low pressure was deleted since this function is not part of the VEGP design. As discussed in VEGP FSAR paragraph 9.2.6.4, each condensate storage tank has sufficient capacity to provide for decay heat removal until the residual heat removal system can be placed in operation.

"Undervoltage RCP Start Turbine Driven Pump" was deleted since this mitigation signal is not part of the VEGP design. See figure 7.2.1-1, sheet 15, of the VEGP FSAR.

Functional Unit 7: The switchover to the containment emergency sump is described as "semi-automatic" because, while the switchover sequence is automatically initiated, certain operator actions are required to complete the sequence. Also, the ECCS takes suction from the containment emergency sump as opposed to the normal sump. Finally, the logic which automatically initiates the switchover sequence does not include a high containment sump level signal. See FSAR subsection 6.3.2 for a complete discussion.

Functional Unit 8: The words "to 4.16 kV ESF Bus" were added to reflect plant-specific design. See FSAR paragraph 8.3.1.1 for discussion of the 4.16 kV busses in the VEGP design.

Functional Unit 9: This functional unit was revised to reflect the fact that P-12 does not perform an ESFAS function and was therefore deleted and P-14 was deleted since this function is covered under a previous functional unit. Also, P-14 is not a 2/3 logic in the VEGP design. See section 7.3 of the VEGP FSAR.

Functional Unit 10: Control room ventilation emergency mode actuation was added to reflect plant-specific design. Table Notation e was added to reflect the fact that, during Modes 5 and 6, if fuel is not being handled, this function should not be required. Action Statement 24 was added to provide an appropriate action for this function and to provide consistency with Specification 3.7.6.

Table Notations: STS Action Statement 16 was deleted since N-1 loop operation will not be permitted.

Action Statement 13 was added to provide an appropriate action for the containment area radiation instrument channels.

Action Statement 15 was revised to provide a time frame for closing the purge valves. This was done for clarification.

Action Statement 19 was revised to make the wording consistent with the remainder of the action statements. The intent is unchanged. The word "annunciator" was deleted since the permissive lights are status lights and are not annunciated.

Action Statement 21 was revised so that it could be applied to the auto-start of auxiliary feedwater on loss of or degraded 4.16 kV ESF bus voltage rather than Action 17. Action 17 requires that the plant be placed in cold shutdown but the Applicable Modes are only Modes 1, 2, and 3. Action 21 requires that the plant be placed in hot shutdown which is more appropriate for the Applicable Modes requirement. The application of Action 21 to other functions listed in Table 3.3-3 is unaffected by this revision because in those cases the total No. of channels requirement is the same as the Minimum Channels Operable requirement.

Specific:

Manual Initiation: Those functions which cannot be manually initiated by system level handswitches have been grouped under safety injection indicating that they are initiated by the safety injection signal.

Initiation Signals and Functions 4, 8, 12, and 15 were deleted since they are not included in the VEGP design. See paragraph 7.3.1.1 of the VEGP FSAR for details.

Reference to phase "B" isolation under initiating signal and Function 5 was deleted reflecting the fact that no equipment is actuated on a phase "B" isolation. See Table 6.2.4-1 of the VEGP FSAR.

Concerning phase "A" Isolation, the response time listed includes only signal processing and condition. The valve closure times are specified under 3/4.6.3.1.

Concerning the loss of power initiating signal, footnotes * and ** were added to provide clarification and to avoid duplicate testing with Specification 3/4.8.1.1.

Table 4.3-2:

General:

Instrument numbers were added to these tables to improve the operator's understanding of Technical Specification compliance.

Cross references to Specification 4.3.3.6 were added for those instruments which also perform a PAMs function.

Specific:

The descriptions of each functional unit, the actuation signals for each functional unit, and the modes for which surveillance is required were revised to be consistent with Table 3.3-3.

3/4.3.3:

See VEGP FSAR subsections 12.3.4, 6.2.4, 7.3.6, 11.5.2, and paragraph 6.2.1.6 for descriptions of the instrumentation listed in Tables 3.3-6 and 4.3-3.

Table 3.3-6 and Table 4.3-3:

General:

These tables have been revised to reflect plant-specific design. See VEGP FSAR section 11.5 for details of the radiation monitoring system.

Since many of the instruments listed in these tables also perform an ESFAS function, references were provided to the appropriate tables in an effort to eliminate duplicate requirements. Where a reference to another table is provided, the more limiting requirement is called for.

Specific:

Functional Unit 3 (STS): The fuel storage pool area monitors are not included in this table on the basis that the VEGP SER analysis of a fuel handling accident does not take credit for the operation of the fuel handling building post-accident exhaust system. See table 15.6 for the VEGP FSAR. Also, the VEGP design does not include criticality monitors. As discussed in VEGP FSAR subsection 12.3.2, an exemption from 10 CFR 70.24 will be filed.

Functional Unit 5 (STS): The plant-specific process monitors are listed in Table 3.3-13.

STS Action Statements 26, 27, and 28 were deleted on the basis that the instruments to which these statements were applied are also covered in Table 3.3-3. Reference to Table 3.3-3 was provided since Specification 3/4.3.2 represents the most limiting requirement.

STS Action Statement 30 was deleted since the process monitors are covered in Table 3.3-13.

Action Statement 27 and Table Notation # were added to provide an appropriate action and applicability for the volume reduction system.

<u>4.3.3.2</u>:

This surveillance requirement was revised for clarification for consistency with the wording provided in the bases for this specification.

<u>3/4.3.3.3</u>:

See subsection 3.7.4 of the VEGP FSAR for a description of seismic instrumentation.

<u>4.3.3.3.2</u>:

The revision to this surveillance requirement is proposed on the basis that certain seismic monitoring instruments are accessible only during MODES 5 or 6. The following instruments are affected:

1) Triaxial time-history accelerograph located at the containment pressurizer support.

2) Triaxial peak accelerographs located at a reactor coolant pump motor and at a steam generator.

These instruments are located in Zone V radiation zones and are therefore inaccessible from the standpoint of ALARA considerations. If the seismic event is less than the operating basis earthquake (OBE) (0.12g for VEGP) and no damage has occurred, the plant can continue operation. However, because of the problem with accessibility, the requirement to perform a channel calibration would necessitate placing the plant in cold shutdown. We feel that this requirement could force the plant into an unnecessary shutdown. Sufficient data will be available from the accessible instruments to allow evaluation of the event. If the OBE has been exceeded, then the data from the peak reading accelerographs will be available since a shutdown will be required. If not, then this data should not be required immediately and a followup report should be sufficient.

Table 4.3-4 has been revised to indicate accessibility as discussed above.

Concerning the interval for performing a channel calibration on actuated instruments, assuming that all the instrumentation could be removed at once and that the required personnel were available, a typical channel calibration of both the Kinemetrics and Engdahl systems would take a minimum of 5 days. If there were any equipment or personnel-related delays this could take even longer.

It is important to calibrate seismic instrumentation soon after a seismic event in order to confirm instrument characteristics and validate data reduction. However, it is also important, and perhaps more important, not to remove the entire system from operation immediately following an earthquake since this is the period of time during which aftershocks, if any, may occur. In fact, it is not unknown for an "aftershock" to actually be a main shock since it is larger than the preceding "foreshock." If all of the seismic instrumentation were taken offline for a channel calibration immediately after an earthquake, important aftershock data would probably be missed.

For the reasons given above, the Technical Specification has been changed from 10 days to 15 days to allow sufficient time for both aftershock recording and channel calibration in phases. Calibration in phases is recommended to allow part of the seismic instrumentation to be online at all times.

Tables 3.3-7 and 4.3-4:

VEGP is equipped with a permanently installed triaxial response spectrum analyzer. In the case of a seismic event, a central controller is actuated by seismic switches. The controller then enables the strong motion accelerometers and their recorders to operate and provides a local alarm light.

After the seismic event is recorded an operator alerted by the local alarm light may remove the tapes and install them one at a time in the magnetic tape playback system.

The playback system has the capability to print out the recorded event on its own graph or send the recorded seismic event to the spectrum analyzer.

The spectrum analyzer compares the actual response spectra (recorded event) with the design response spectra (programmed). During the analyzing process the spectrum analyzer has the capability to alert the operator of an event greater than the design response spectra with an indicating light. This capability allows the operator the option to print out only those responses which indicate a possible event.

The analog channel operational test is not applicable to this device on the basis that the analyzer does not provide a signal for an alarm, interlock, and/or trip function.

This system achieves the intent of Regulatory Guide 1.12, Revision 1.

3/4.3.3.5:

This specification was revised so that the transfer switches and control circuits do not have to be listed in Table 3.3-9. Such a listing would be extensive and would not provide any useful information to the operators. The required transfer switches and control circuits will be listed in the procedures associated with the remote shutdown panels. The instrumentation listed in Table 3.3-9 reflects the minimum instrumentation required to achieve and/or maintain safe shutdown. The instrumentation which is isolated from the control room meets the recommendations of CMEB 9.5.1, while the redundancy in the A and B shutdown panels meets the intent of GDC 19 of 10 CFR 50.

The limiting conditions for operation and surveillance requirements were rewritten to qualitatively specify the components which require transfer and/or control circuits to maintain independence from the control room so that safe shutdown can be achieved in the event of a control room fire. See VEGP FSAR subsection 7.4.3 for a discussion of the systems required for safe shutdown and safe shutdown from outside the control room. References to power circuits were deleted on the basis that power for the required components is independent of the control room and shutdown panels.

3/4.3.3.6 and Table 3.3-10:

This specification and table have been revised to be consistent with the plant-specific hardware which has been installed in accordance with Regulatory Guide 1.97. Additionally, the recommendation that only Category 1 variables be included in Technical Specifications per NRC memorandum for Darrell Eisenhut from Roger Mattson, dated October 11, 1983, and entitled, "Technical Specifications for Post Accident Monitoring Instrumentation," has been incorporated. (See paragraph 7.5.2.3 of the VEGP FSAR.) Some of the specific changes to the Technical Specifications are discussed in more detail in the following paragraphs.

Action statements were incorporated into Table 3.3-10 due to the increased complexity of the actions resulting from an increase in the number of required channels for some instruments. The revised actions are in no case less restrictive than the actions contained in the generic STS.

The required number of channels for some instruments has been increased from two to three to be consistent with Regulatory Guide 1.97.

New Action 28.a was added to allow credit for the increased number of required channels from two to three instruments. Generic STS only requires two channels. The VEGP Technical Specifications requires three channels from some instruments. In recognition of this increased capability, a 30-day action is stipulated for loss of a single channel. Operation with only one channel operable is restricted to 7 days and is consistent with the generic STS.

Action Statement 31 was added to permit utilization of a temporary alternate means of monitoring steam line radiation or containment radiation. This would provide the required indication while repairs are underway on the permanent monitors. The necessary data would still be available and a plant shutdown might be averter. Action Statment 32 and footnote * to Action Statment 32 were added due to the historical difficulties which have been experienced with the RVLIS. This action would apply to the first fuel cycle only. Both of these proposed revisions have been approved by the NRC for the Diablo Canyon Technical Specification.

3.3.3.7:

The revision to the applicability is based on the fact that during MODES 5 and 6, the plant is in a stable, safe shutdown condition. As long as movement of irradiated fuel or loads over irradiated fuel is not taking place, there is not a potential for a radioactive release and, thus, the plant will remain in a safe shutdown condition. Under these conditions it is necessary to require the operability of the chlorine detection system to isolate the control room.

The word "recirculation" was replaced with "isolation" for the sake of consistency in plant-specific terminology. The term "isolation" is descriptive of the mode of operation of the control room HVAC system in the event of a toxic gas release.

The alarm/trip setpoints of the chlorine detectors must be adjusted to actuate with a chlorine concentration of $\leq$ 2 ppm to be consistent with the VEGP toxic gas analysis. See subsections 6.4.2, 6.4.3, 6.4.4, and 7.3.6 of the VEGP FSAR.

3/4.3.3.8:

We are proposing to delete this specification and the specifications for fire suppression systems (3/4.7.11), fire rated assemblies (3/4.7.12), and the fire brigade (6.2.2 (e)), on the basis that the VEGP Fire Protection Program will be completely described and controlled through the FSAR rather than the combination of the FSAR and the Technical Specifications. This is the same concept which has been adopted by another utility. This will, in effect, transfer Fire Protection Program commitments, reporting requirements, and amendments from the jurisdiction of 10 CFR 50.73 and 10 CFR 50.90 to 10 CFR 50.59 and 10 CFR 50.71(e).

The administrative portion of the Technical Specifications will require that the plant review board (PRB) be responsible for the review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the General Manager-Vogtle Nuclear Operations (6.5.1.6(n)). In addition, Specification 6.5.2.8 will require an audit of the Fire Protection Program and implementing procedures at least once per 24 months and an independent fire protection and loss presention inspection and an audit at least once per 12 months. Specification 6.8.1 will require that written procedures be established, implemented, and maintained for the Fire Protection Program.

Surveillance requirements, LCOs, and actions to be taken for the detection and suppression systems and the fire-rated assemblies will be in accordance with VEGP commitments to the NFPA Codes and Standards and NML Standards as they appear in the VEGP FSAR.

Table 3.3-11:

General:

See paragraph 11.5.2.3 of the VEGP FSAR for a description of the PERMS.

Plant-specific instrument numbers were added for clarity and to assist operators in ensuring compliance with the Technical Specifications.

Action statement numbers were revised to reflect renumbering to maintain continuity of action statements in Table Notation.

Item 1.d.: The control building sumps effluent line was added to reflect plant design.

Item 2.a.: The additional description reflects plant-specific nomenclature.

Item 2.b.: The deletion of the component cooling water system reflects a plant-specific design where any system leakage would be into the nuclear service cooling water system.

Items 3.a. & b.: The continuous composite sampler and sampler flow monitor were deleted to reflect current plant-specific design. Their use is optional if items 1.b. and 1.c. are utilized.

Item 4.c.: The entry for "Discharge Canal" was revised to "Flow to Blowdown Sump" to reflect the plant design.

Items 5.a & b and Footnote*: The deletion of radioactivity recorders was made to reflect plant-specific design. VEGP has no recorders that provide alarm/trip functions.

Action 39 (STS): This action was deleted since it applies only to item 5.b. which was previously deleted.

Table 4.3-8:

General:

The changes and associated justifications noted for Table 3.3-11 are also applicable to this table.

Table 3.3-12:

General:

See paragraph 11.5.2.3 of the VEGP FSAR for a description of the PERMS.

Plant-specific instrument numbers were added for clarity.

ACTION statements were renumbered consecutively.

Specific:

Item 1. The terminology change (from waste gas holdup system to gaseous waste processing system) was made to reflect plant-specific nomenclature.

Items 1.b., c., & e.: The deletions reflect plant-specific design. The WGPS exhausts via the plant vent, so the instruments provided for the plant vent provide the Appendix I requirements.

Item 2.A The deletions reflect plant-specific design which are not applicable to VEGP.

Item 2.B These revisions were made to reflect the specific instrumentation provided for the recombiners.

Item 3. The addition of "Air Ejector and Steam Packing Exhauster" reflects plant-specific nomenclature.

Item 4. The inclusion of "Plant Vent" reflects plant-specific nomenclature.

Items 5., 6., & 7: The deletions of these items reflect plant-specific design where each system is directed to the "Plant Vent."

Item 8. The addition of "Solidification Building" identifies a plant-specific design and instrumentation.

Item 9. The system deletion reflects a plant-specific design.

Table Notation ** was revised to incorporate the words "Gaseous Waste Processing System" to reflect plant-specific nomenclature.

Table Notation *** was added on the basis that the instrumentation to which it is applied is not required unless the associated systems are in operation. Also, the plant vent is in the final release point for the GWPS.

Action Statements 48 & 49 (STS): The deletion of Actions 48 and 49 resulted from the deletion of instruments which are not applicable to VEGP.

Action Statements 50 and 52 (STS): Actions Statements 50 and 52 were replaced with Action Statements 40 and 42 in items 2(a) and 2(B) of Table 3.3-12 (hydrogen and oxygen monitors, respectively). Action Statements 40 and 42 incorporate the requirements of Action Statements 50 and 52 to the VEGP specific design as follows:

Action Statement 40(a): The large mass of gas in the gaseous waste processing system (GWPS) would prevent any rapid increase in the oxygen concentration. The periodic grab samples would detect any buildup of oxygen and the inlet oxygen monitor would alarm if an oxygen concentration of 3 percent by volume is reached for the combined stream of the oxygen supply, the recirculating gases, and the inlet to the GWPS.

Action Statement 40(b): Any buildup of the H and O would be detected and alarmed by the inlet hydrogen monitor and the outlet oxygen monitor. If the O concentration continued to increase, the high setpoint would isolate the O supply.

Action Statement 40(c): The large mass of gas in the GWPS would prevent any rapid increase in the O concentration. The periodic grab samples would detect any buildup of oxygen.

Action Statement 42: The large mass of gas in the GWPS would prevent any rapid increase in the hydrogen concentration. The period grab samples would monitor the buildup of hydrogen.

Table 4.3-9:

General:

The changes and associated justifications noted for Table 3.3-12 are also applicable to this table.

The addition of the words "ANALOG" and "OPERATIONAL" were made to bring NUREG 0472 terminology in line with NUREG 0452.

Specific:

Items 3d, 3e, 4d, 4e, 5d, and 5e: A new note 5 was added which indicates that a channel check needs to be made only when a release is under way. This reflects the fact that there may be times when releases are not being made via these pathways.

Table Notations 4 & Old Note 5: This revision was made to reflect the fact that "manufacturer's recommendations" may differ from STS requirements, and since the manufacturer's recommendations would be more appropriate in achieving proper calibration, the intent of the specification is preserved.

3/4.3.3.11:

This specification was added to ensure probability of the electric steam boiler isolation, steam generator blowdown isolation, and letdown line isolation functions as discussed in section 7.6 of the VEGP FSAR.

3/4.3.4:

The proposed revision to this specification was made to incorporate recently established criteria for the operability of turbine overspeed protection as presented in 3.5.1.3 of the VEGP SER. See also paragraph 10.2.3.6 and response to Question 430.50 of the VEGP FSAR.

A similar specification has already been approved and implemented in the Callaway Unit 1 Technical Specifications.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All reactor coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant loops in operation, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

*See Special Test Exceptions Specification 3.10.4.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 At least two of the reactor coolant loops listed below shall be OPERABLE with at least two reactor coolant loops in operation when the Reactor Trip System breakers are closed and at least one reactor coolant loop in operation when the Reactor Trip System breakers are open:*

a. Reactor Coolant Loop ~~[A]~~ 1 and its associated steam generator and reactor coolant pump,

b. Reactor Coolant Loop ~~[B]~~ 2 and its associated steam generator and reactor coolant pump,

c. Reactor Coolant Loop ~~[C]~~ 3 and its associated steam generator and reactor coolant pump, and

d. Reactor Coolant Loop ~~[D]~~ 4 and its associated steam generator and reactor coolant pump.

APPLICABILITY: MODE 3.**

ACTION:

a. With less than the above required reactor coolant loops OPERABLE, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With only one reactor coolant loop in operation and the Reactor Trip System breakers in the closed position, within 1 hour open the Reactor Trip System breakers.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required reactor coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~[17%]~~ , at least once per 12 hours. of wide range (LT-0501, LT-0502, LT-0503, LT-0504)

4.4.1.2.3 The required reactor coolant loops shall be verified in operation and circulating reactor coolant at least once per 12 hours.

---

*All reactor coolant pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** See Special Test Exception Specification 3.10.4b.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the ~~loops~~ loops/trains listed below shall be OPERABLE and at least one of these ~~loops~~ loops/trains shall be in operation:*

a. Reactor Coolant Loop ~~[A]~~ 1 and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop ~~[B]~~ 2 and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop ~~[C]~~ 3 and its associated steam generator and reactor coolant pump,**

d. Reactor Coolant Loop ~~[D]~~ 4 and its associated steam generator and reactor coolant pump,**

e. RHR ~~Loop~~ Train [A], and

f. RHR ~~Loop~~ Train [B].

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required ~~loops~~ loops/trains OPERABLE, immediately initiate corrective action to return the required ~~loops~~ loop/train to OPERABLE status as soon as possible; if the remaining OPERABLE ~~loop~~ loop/train is an RHR ~~loop~~ train, be in COLD SHUTDOWN within 24 hours.

b. With no ~~loop~~ loop/train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop/train to operation.

---

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started ~~with one or more of the Reactor Coolant System cold leg temperatures less than or equal to [275]°F unless~~ unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required reactor coolant pump(s), if not in operation, shall be determined OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to ~~[17]~~% at least once per 12 hours. of wide range (LT-0501, LT-0502, LT-0503, LT-0504)

4.4.1.3.3 At least one reactor coolant or RHR ~~loop~~ train shall be verified in operation and circulating reactor coolant at least once per 12 hours.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) ~~loop~~ train shall be OPERABLE and in operation*, and either:

a. One additional RHR ~~loop~~ train shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than [17]% of wide range (LT-0501, LT-0502, LT-0503, LT-0504).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR ~~loops~~ trains inoperable or with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR ~~loop~~ to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR ~~loop~~ train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR ~~loop~~ train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR ~~loop~~ train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

---

*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR ~~loop~~ train may be inoperable for up to 2 hours for surveillance testing provided the other RHR ~~loop~~ train is OPERABLE and in operation.

***A reactor coolant pump shall not be started ~~with one or more of the Reactor Coolant System cold leg temperatures less than or equal to [275]°F~~ unless the secondary water temperature of each steam generator is less than 50 °F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two residual heat removal (RHR) ~~loops~~ trains shall be OPERABLE* and at least one RHR ~~loop~~ train shall be in operation.**

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required RHR ~~loops~~ trains OPERABLE, immediately initiate corrective action to return the required RHR ~~loops~~ trains to OPERABLE status as soon as possible.

b. With no RHR ~~loop~~ train in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR ~~loop~~ train to operation.

c. With the Reactor Makeup Water Storage Tank (RMWST) discharge valves (1208-U4-175, 1208-U4-176, 1208-U4-177 and 1208-U4-183) not closed and secured in position, immediately close and secure in position the RMWST discharge valves.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2.1 At least one RHR ~~loop~~ train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

4.4.1.4.2.2 Valves 1208-U4-175, 1208-U4-177, 1208-U4-185, and 1208-U4-176 shall be verified closed and secured in position by mechanical stops at least once per 31 days.

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*One RHR ~~loop~~ train may be inoperable for up to 2 hours for surveillance testing provided the other RHR ~~loop~~ train is OPERABLE and in operation.

**The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

ISOLATED LOOP (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.4.1.5 The boron concentration of an isolated loop shall be maintained greater than or equal to the boron concentration of the operating loops.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

With the requirements of the above specification not satisfied, do not open the isolated loop's stop valves; either increase the boron concentration of the isolated loop to within the limits within 4 hours or be in at least HOT STANDBY within the next 6 hours with the unisolated portion of the RCS borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F.

SURVEILLANCE REQUIREMENTS

4.4.1.5 The boron concentration of an isolated loop shall be determined to be greater than or equal to the boron concentration of the operating loops at least once per 24 hours and within 30 minutes prior to opening either the hot leg or cold leg stop valves of an isolated loop.

REACTOR COOLANT SYSTEM

ISOLATED LOOP STARTUP [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.4.1.6 A reactor coolant loop shall remain isolated until:

a. The isolated loop has been operating on a recirculation flow of greater than or equal to ___ gpm for at least 90 minutes and the temperature at the cold leg of the isolated loop is within 20°F of the highest cold leg temperature of the operating loops, and

b. The reactor is subcritical by at least 1% $\Delta k/k$.

APPLICABILITY: All MODES.

ACTION:

With the requirements of the above Specification not satisfied, suspend startup of the isolated loop.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The isolated loop cold leg temperature shall be determined to be within 20°F of the highest cold leg temperature of the operating loops within 30 minutes prior to opening the cold leg stop valve.

4.4.1.6.2 The reactor shall be determined to be subcritical by at least 1% $\Delta k/k$ within 30 minutes prior to opening the cold leg stop valve.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, ~~immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.~~ either comply with the provisions of Specification 3.4.9.3 or ensure pressure relief capability through an RHR loop.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer Code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The (Later) pressurizer shall (LT-0459, LT-0460, LT-0461) be OPERABLE with a water volume of less than or equal to ~~[1636]~~ cubic feet, and at least two groups of pressurizer heaters each having a capacity of at least {150} kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With only one group of pressurizer heaters OPERABLE, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the Reactor Trip System breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified by energizing the heaters and measuring circuit current at least once per 92 days.

~~4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.~~

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD SHUTDOWN~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, 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, and

   1. With only one Class 1 PORV OPERABLE, restore at least a total of two Class 1 PORVs to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours, or

   2. With no Class 1 PORVs OPERABLE, restore at least two Class 1 PORVs to OPERABLE status within ~~1~~ 4 hours or be in HOT STANDBY within the next 6 hours and ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour (1) restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its assocated solenoid valve; and (2) apply ACTION b above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 ~~In addition to the requirements of Specification 4.0.5,~~ Each PORV shall be domonstrated OPERABLE at least once per 18 months by:

a. Operating the valve through one complete cycle of full travel, and

b. Performing a CHANNEL CALIBRATION. ~~of the actuation instrumentation.~~

3/4.4.4 RELIEF VALVES

SURVEILLANCE REQUIREMENTS

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION a and b in Specification 3.4.4.

~~4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:~~

~~a. Manually transferring motive and control power from the normal to the emergency power supply, and~~

~~b. Operating the valves through a complete cycle of full travel.~~

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

Loop 1 TE 413A & B
Loop 2 TE 423A & B
Loop 3 TE 433A & B
Loop 4 TE 443A & B

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program. ~~and the requirements of Specification 4.0.5~~.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1) All nonplugged tubes that previously had detectable wall penetrations ~~(greater than 20%)~~.

2) Tubes in those areas where experience has indicated potential problems, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1) The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2) The inspections include those portions of the tubes where imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to ~~at least once per 20~~ 12 to 24 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

   4) A Condition IV main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this specification:

1) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

2) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

3) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;

5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be repaired or plugged ~~removed from service~~ and is equal to ~~[~~40~~]~~%~~*~~ of the nominal tube wall thickness;

7) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.4.5.3c., above;

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

~~*Value to be determined in accordance with the recommendations of Regulatory Guide 1.121, August 1976.~~

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be repair or determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes repaired or plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1) Number and extent of tubes inspected,

2) Location and percent of wall-thickness penetration for each indication of an imperfection, and

3) Identification of tubes repaired or plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR Part ~~50.72~~ 50.73 and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE

INSPECTED DURING INSERVICE INSPECTION

~~Preservice Inspection~~	~~No~~			~~Yes~~		
No. of Steam Generators per Unit	~~Two~~	~~Three~~	~~Four~~	~~Two~~	~~Three~~	Four
First Inservice Inspection	~~All~~			~~One~~	~~Two~~	Two
Second & Subsequent Inservice Inspections	~~One~~[1]			~~One~~[1]	~~One~~[2]	One~~3~~ 1

~~Table Notation:~~

1. ~~The inservice inspection~~ may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

1. ~~3.~~ Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections ~~shall follow the instructions described in 1 above.~~

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Repair or plug ~~Plug~~ defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Repair or plug ~~Plug~~ defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Repair or plug ~~Plug~~ defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., repair or plug ~~plug~~ defective tubes and inspect 2S tubes in each other S. G. ~~Prompt notification to NRC pursuant to specification 6.9.1~~	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and repair or plug ~~plug~~ defective tubes. ~~Prompt notification to NRC pursuant to specification 6.9.1~~	N/A	N/A

$S = 3 \frac{N}{n} \%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

Notification to NRC pursuant to 10 CFR 50.73.

Per Generic Letter 83-43

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere [Gaseous or Particulate] Radioactivity Monitoring System,

b. The Containment Normal ~~Pocket~~ Sumps Level, Reactor Cavity Sump Level, and Flow Monitoring System, and

c. Either the [containment air cooler condensate flow rate] or a Containment Atmosphere [Gaseous or Particulate] Radioactivity Monitoring System, not taken credit for in 3.4.6.1 a above.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With only two of the above required Leakage Detection Systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required Gaseous or Particulate Radioactive Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

a. Containment Atmosphere Gaseous ~~and~~ and/or Particulate Monitoring Systems-performance of CHANNEL CHECK, CHANNEL CALIBRATION, and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3,

b. Containment Normal ~~Pocket~~ Sumps Level, Reactor Cavity Sump Level, and Flow Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months, and

~~c. [Specify appropriate surveillance tests depending upon the type of Leakage Detection System selected.]~~

c. Containment Air Cooler Condensate Monitoring System - performance of CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary ~~reactor~~-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and {500} gallons per day through any one steam generator ~~not isolated from the Reactor Coolant System~~,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

~~e. GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig~~.

e. ~~f.~~ 0.5 GPM leakage per nominal inch of valve size up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve. ~~specified in Table 3.4-1~~. The maximum allowable leakage of any Reactor Coolant System Pressure Isolation valves shall be as specified in Table 3.4-1.*

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the ~~above~~ limit specified in Table 3.4-1, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

* Test pressures less than 2235 psig but greater than 350 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere [gaseous or particulate] radioactivity monitor at least once per 12 hours;

b. Monitoring the containment ~~pocket~~ normal sump~~s~~ and reactor cavity sump inventory and discharge at least once per 12 hours;

~~c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;~~

c. ~~d.~~ Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and

d. ~~e.~~ Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

Replace with insert to page 3/4 4-21.

~~a. At least once per 18 months,~~

~~b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,~~

~~c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve, and~~

~~d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.~~

a. ~~e.~~ As outlined in the ASME Code, Section XI, paragraph IWV-3427(b).

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

Insert to page 3/4 4-21

a. At least once during each refueling outage (Leak testing should be performed after all disturbances to the valves are complete, such as before reaching power operation following a refueling outage);

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve (Leak testing should be performed after all disturbances to the valves are complete);

c. For systems rated at less than 50% of RCS design pressure, each time the valve has moved from its fully closed position unless justification is given;

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

	VALVE NUMBER	VALVE SIZE(in.)	FUNCTION	MAXIMUM ALLOWABLE LEAKAGE(gpm)
1.	HV-8701A	12	RHR Suction (gate valve)	5.0
2.	HV-8701B	12	RHR Suction (gate valve)	5.0
3.	HV-8702A	12	RHR Suction (gate valve)	5.0
4.	HV-8702B	12	RHR Suction (gate valve)	5.0
5.	1204-U4-120	2	SI-Hot Leg 2nd Isolation Valve	1.0
6.	1204-U4-121	2	SI-Hot Leg 2nd Isolation Valve	1.0
7.	1204-U4-122	2	SI-Hot Leg 2nd Isolation Valve	1.0
8.	1204-U4-123	2	SI-Hot Leg 2nd Isolation Valve	1.0
9.	1204-U6-079	10	Accumulator 2nd Isolation Valve	5.0
10.	1204-U6-080	10	Accumulator 2nd Isolation Valve	5.0
11.	1204-U6-081	10	Accumulator 2nd Isolation Valve	5.0
12.	1204-U6-082	10	Accumulator 2nd Isolation Valve	5.0
13.	1204-U6-083	10	Injection Line 1st Isolation Valve	5.0
14.	1204-U6-084	10	Injection Line 1st Isolation Valve	5.0
15.	1204-U6-085	10	Injection Line 1st Isolation Valve	5.0
16.	1204-U6-086	10	Injection Line 1st Isolation Valve	5.0
17.	1204-U6-124	6	SI-Hot Leg 1st Isolation Valve	3.0
18.	1204-U6-125	6	SI-Hot Leg 1st Isolation Valve	3.0
19.	1204-U6-126	6	SI-Hot Leg 1st Isolation Valve	3.0
20.	1204-U6-127	6	SI-Hot Leg 1st Isolation Valve	3.0
21.	1204-U6-128	8	RHR-Hot Leg 2nd Isolation Valve	4.0
22.	1204-U6-129	8	RHR-Hot Leg 2nd Isolation Valve	4.0
23.	1204-U6-143	2	SI-Cold Leg 2nd Isolation Valve	1.0
24.	1204-U6-144	2	SI-Cold Leg 2nd Isolation Valve	1.0
25.	1204-U6-145	2	SI-Cold Leg 2nd Isolation Valve	1.0
26.	1204-U6-146	2	SI-Cold Leg 2nd Isolation Valve	1.0
27.	1204-U6-147	6	RHR Cold Leg 2nd Isolation Valve	3.0
28.	1204-U6-148	6	RHR Cold Leg 2nd Isolation Valve	3.0
29.	1204-U6-149	6	RHR Cold Leg 2nd Isolation Valve	3.0
30.	1204-U6-150	6	RHR Cold Leg 2nd Isolation Valve	3.0

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady-State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

PARAMETER	STEADY-STATE LIMIT	TRANSIENT LIMIT
Dissolved Oxygen*	≤ 0.10 ppm	≤ 1.00 ppm
Chloride	≤ 0.15 ppm	≤ 1.50 ppm
Fluoride	≤ 0.15 ppm	≤ 1.50 ppm

*Limit not applicable with T_{avg} less than or equal to 250°F.

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
Dissolved Oxygen*	At least once per 72 hours
Chloride	At least once per 72 hours
Fluoride	At least once per 72 hours

*Not required with T_{avg} less than or equal to 250°F

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to $100/\bar{E}$ microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than $100/\bar{E}$ microCuries per gram, of gross radioactivity be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the reactor coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

FIGURE 3.4-1

DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1 μCi/gram DOSE EQUIVALENT I-131

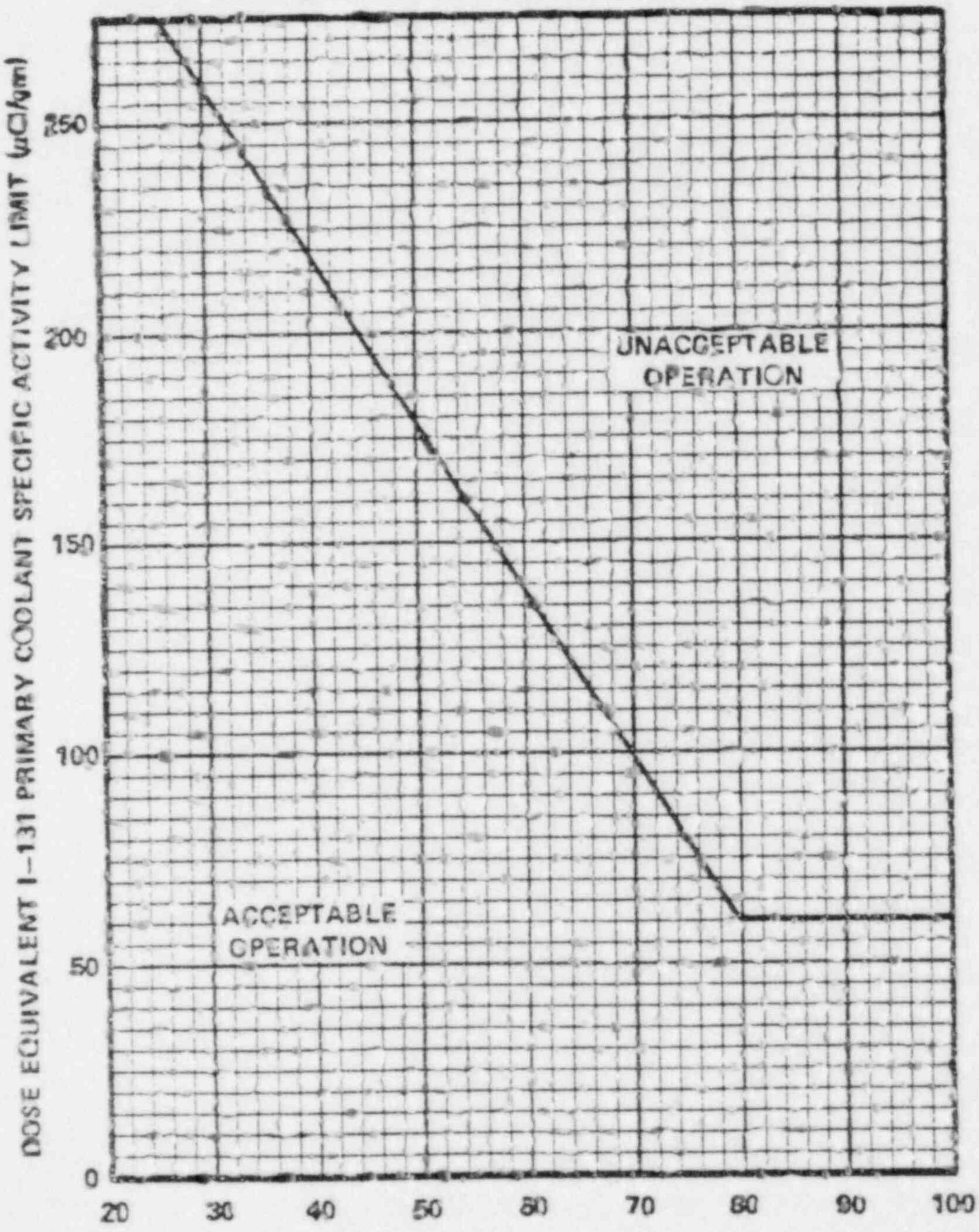


FIGURE 3.4-1

DOSE EQUIVALENT I–131 ~~Primary~~ Reactor Coolant Specific Activity Limit Versus Percent of RATED THERMAL POWER with the ~~Primary~~ Reactor Coolant Specific Activity > 1.0 μCi/gram Dose Equivalent I-131

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination*	At least once per 72 hours.	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
3. Radiochemical for $\bar{E}$ Determination**	1 per 6 months***	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or $100/\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a ~~THERMAL POWER change~~ ~~exceeding~~ continuous increase or decrease in THERMAL POWER that exceeds 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

TABLE 4.4-4 (Continued)

14 TABLE NOTATIONS

*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than ~~10~~ 14 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available, isotopic decay data may be used for pure beta-emitting radionuclides.

**A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than ~~10~~ 14 minutes and all radioiodines, which ~~is~~ are identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

***Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the Reactor Coolant System is restored within its limits.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of [100]°F in any 1-hour period,

b. A maximum cooldown of [100]°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to [10]°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

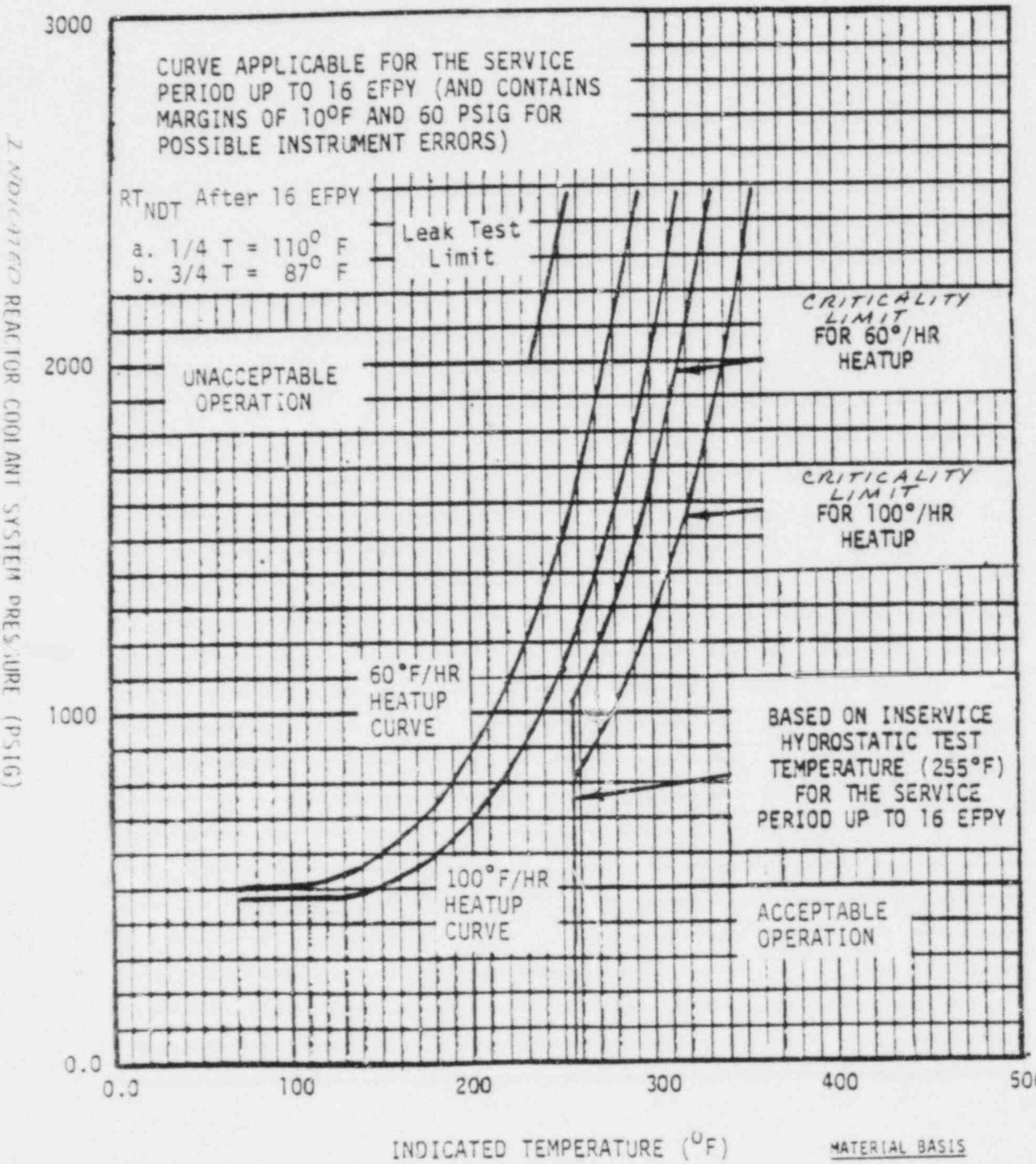
SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per ~~30 minutes~~ hour during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

FIGURE 3.4-2

Reactor Coolant System Pressure-Temperature Limits
Versus $60^{o}F/HR$ and $100^{o}F/HR$ Heatup Rate
And Hydrostatic Test Limit



MATERIAL BASIS

Cupper Content: Assumed - .10 Wt %
(Actual - .06 Wt %)

RT_{NDT} Initial: Assumed - 40°F
(Actual - 30°F)

RT_{NDT} After 16 EFPY @ 1/4T = 110°F
@ 3/4T = 87°F

FIGURE 3.4-2

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO ____ EFPY

FIGURE 3.4-3

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO ______ EFPY

FIGURE 3.4-3

Reactor Coolant System Pressure-Temperature Limits Versus Cooldown Rates

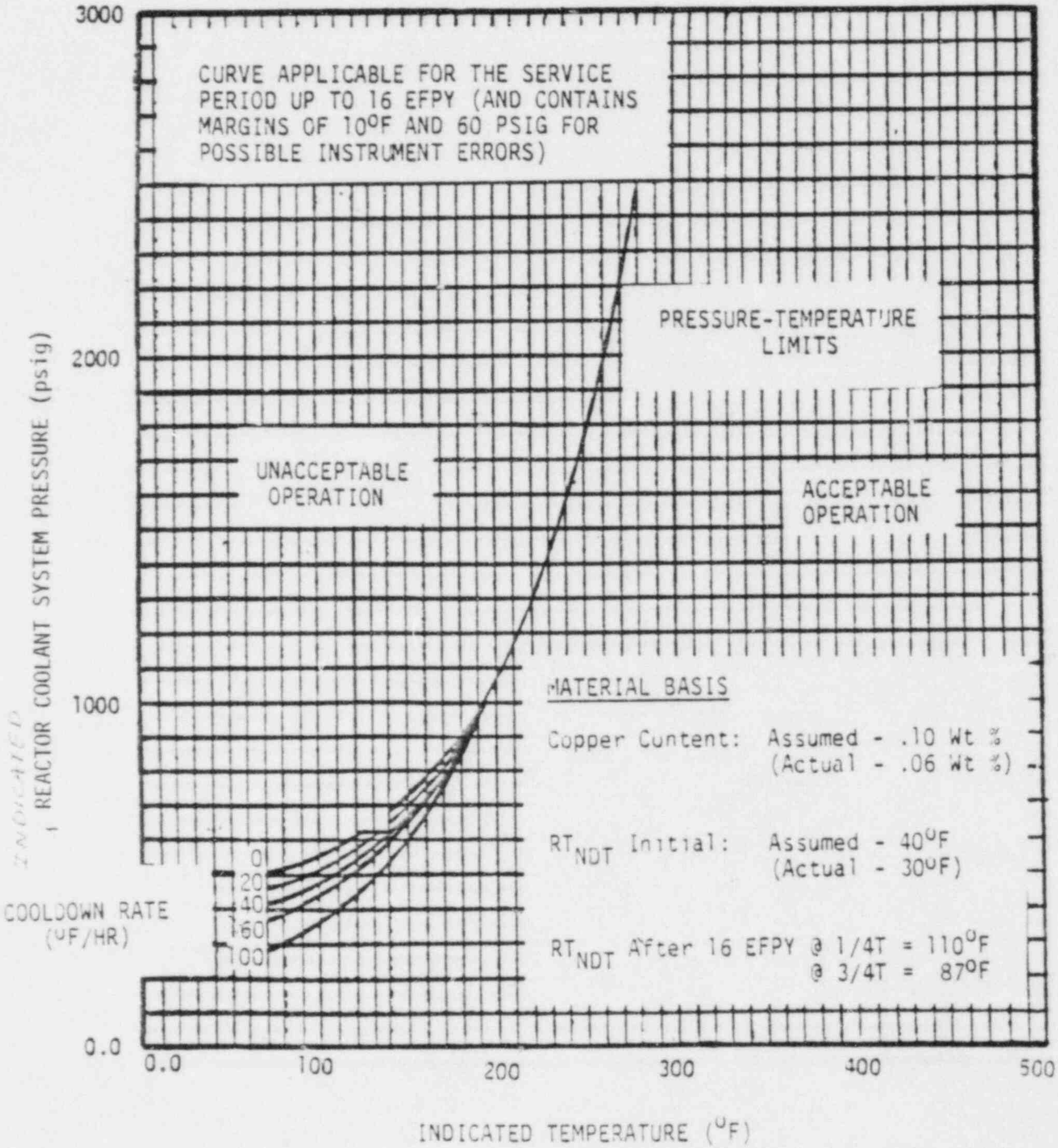


TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)

~~W-STS~~ VOGTLE-UNITS

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	58.5	4.00	1st Refueling
Y	241.0	3.69	5
V	61.0	3.69	9
X	238.5	4.00	15
W	121.5	4.00	Standby
Z	301.5	4.00	Standby

~~NOV 2 1981~~

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of [100]°F in any 1-hour period,

b. A maximum cooldown of [200]°F in any 1-hour period, and

c. A maximum auxiliary spray water temperature differential of ~~[320]~~ 625°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per hour ~~30 minutes~~ during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

REACTOR COOLANT SYSTEM

COLD OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Cold Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with ~~a~~ lift settings ~~of less than or equal to [450] psig, or~~ which vary with RCS temperature and which do not exceed the limits established in Figure 3.4-4, or

c. ~~b.~~ The Reactor Coolant System (RCS) depressurized with an RCS vent ~~of greater than or equal to ____ square inches.~~ capable of relieving at least 670 gpm water flow at 470 psig.

APPLICABILITY: MODES 4, ~~when the temperature of any RCS cold leg is less than or equal to [275]°F, MODE 5 and MODE 6~~, 5 and 6, with the reactor vessel head on.

ACTION:

a. With one PORV and one RHR suction relief valve inoperable, either restore ~~the inoperable~~ two PORVs to OPERABLE or two RHR suction relief valves status within 7 days or depressurize and vent the RCS ~~through at least a ____ square inch vent~~ as specified in 3.4.9.3.C, above within the next 8 hours.

b. With both PORVs and both RHR suction relief valves inoperable, depressurize and vent the RCS ~~through at least a ____ square inch vent~~ as specified in 3.4.9.3.C above within 8 hours.

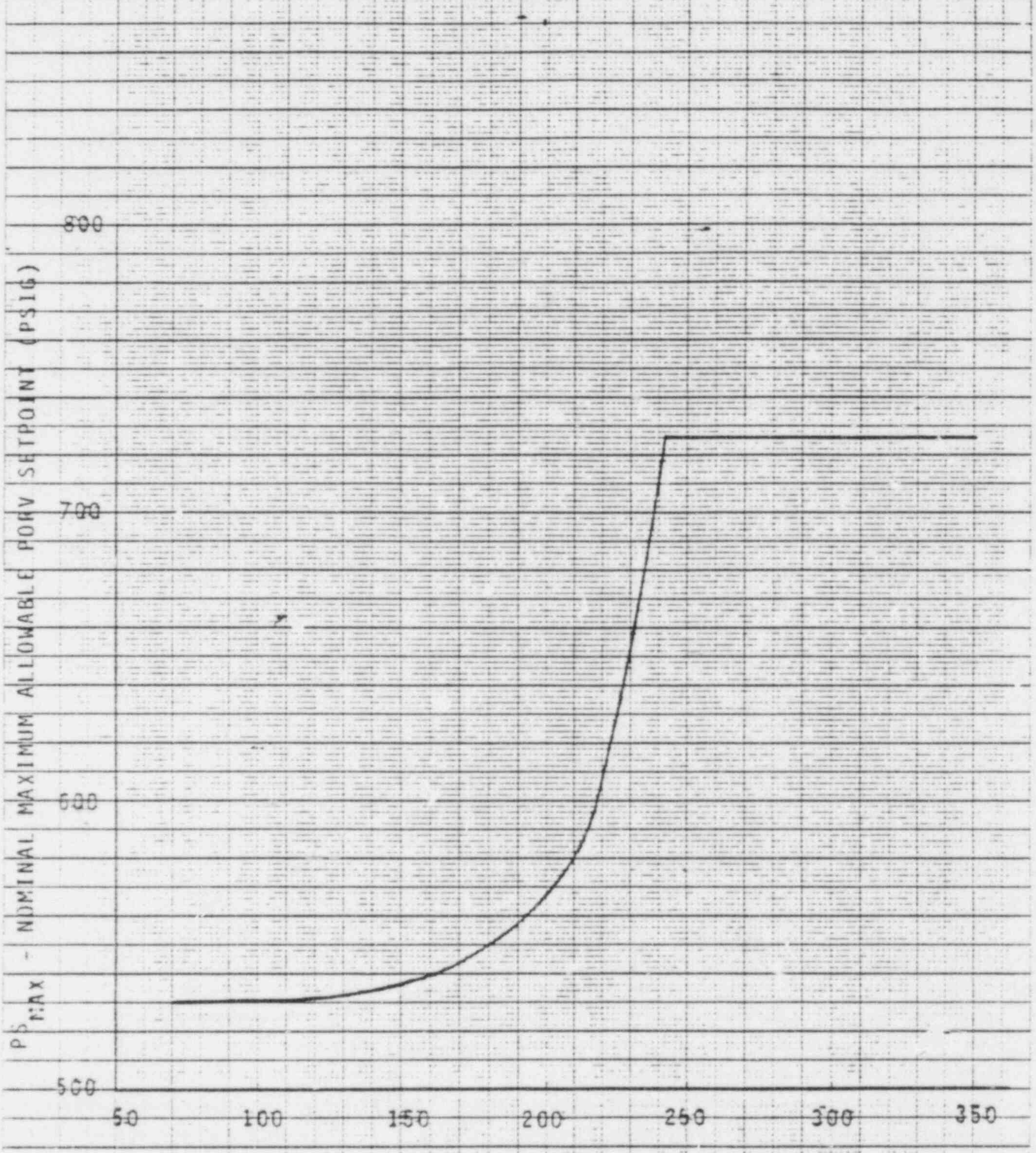
c. In the event either the PORVs, or the RHR suction relief valves, or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

b. Two residual heat removal (RHR) suction relief valves each with a setpoint of 450 psig ± 1%, or

FIGURE 3.4 - 4

MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT
FOR THE COLD OVERPRESSURE PROTECTION SYSTEM



REACTOR COOLANT SYSTEM

COLD OVERPRESSURE PROTECTION SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.~~2~~3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve PSV-8708A by verifying that RHR RCS suction isolation valves HV-8701A and HV-8701B are open at least once per 12 hours.

b. For RHR suction relief valve PSV-8708B by verifying that RHR RCS suction isolation valve HV-8702A and HV-8702B are open at least once per 12 hours.

c. Testing pursuant to Specification 4.0.5.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

See insert for page 3/4 4-38.

~~4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

INSERT TO PAGE 3/4 4-38

4.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be demonstrated pursuant to Specification 4.0.5.

4.4.10.2 In addition to the requirements of Specification 4.0.5, the Reactor Coolant Pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14 Revision 1, August 1975.

4.4.10.3 In addition to the requirements of Specification 4.0.5 the four main steamlines and feedwater lines from the containment penetration flued head outboard weld, to the upstream weld of the five-way restraint, which is downstream of the main steam isolation valves, shall be inspected. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical.

REACTOR COOLANT SYSTEM

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 ~~At least one~~ Two Reactor ~~Coolant System~~ vessel head vent paths, each consisting of [two] vent valves(s) ~~and [one] block valve~~ powered from emergency busses, shall be OPERABLE and closed. ~~at each of the following locations:~~

~~a. [Reactor vessel head],~~

~~b. [Pressurizer steam space], and~~

~~c. [Reactor Coolant System high point].~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor ~~Coolant System~~ vessel head vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves ~~and block valves~~ in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two ~~or more~~ Reactor ~~Coolant System~~ vessel head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the vent valves ~~and block valves~~ in the inoperable vent paths, and restore at least ~~[two]~~ one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

~~4.4.11.1 Each Reactor Coolant System vent path block valve not required to be closed by ACTION a. or b., above, shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel from the control room.~~

4.4.11.~~2~~ Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each vent valve through at least one complete cycle of full travel from the control room, and

c. Verifying flow through the Reactor Coolant System vent paths during venting.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.4

3/4.4:

Instrument tag numbers were added, as appropriate, to assist the operator in ensuring compliance with these Technical Specifications.

3.4.1.2:

The applicability has been superscripted with ** to indicate that special test exception 3.10.4.b allows the suspension of 3.4.1.2 during hot rod drop time tests in Mode 3 provided at least two reactor coolant loops are operable. This change has already been approved and implemented in the Byron Unit 1 Technical Specifications.

4.4.1.2.2:

The words "of wide range" were inserted for clarification.

3.4.1.3:

This specification was revised to utilize the VEGP specific terminology "Reactor Coolant Loop" and "RHR Train."

The restriction on cold leg temperature in footnote ** was deleted since the COPS arming temperature of less than or equal to 350°F includes all of Mode 4. See paragraphs 5.2.2.10 and 5.2.2.11 of the VEGP FSAR.

4.4.1.3.2:

The words "of wide range" were inserted for clarification.

4.4.1.3.3:

The word "loop" was replaced with "train" for the sake of consistency in terminology.

3.4.1.4.1:

The word "loop" was replaced with "train" for consistency with VEGP terminology.

Footnote *** was modified to delete restriction on cold leg temperature since the COPS arming temperature of less than or equal to 350°F includes all of Mode 4. See paragraphs 5.2.2.10 and 5.2.2.11 of the VEGP FSAR.

The words "of wide range" were inserted into 3.4.1.4.1.b for clarification.

4.4.1.4.1.2:

The word "loop" was replaced with "train" for consistency with VEGP terminology.

3/4.4.1.4.2:

This specification was revised to utilize the VEGP specific terminology "Reactor Coolant Loop" and "RHR Train."

3/4.4.1.4.2.2:

This specification was revised to preclude the possibility of a boron dilution accident by isolating the sources of unborated water. These requirements are consistent with the assumed initial conditions of the boron dilution accident analyses. See the response to Confirmatory Item 47, Bailey to Denton, GN-666, dated 7/25/85.

3/4.4.1.5 (STS):

The VEGP design does not include RCS loop isolation valves. Therefore, this specification is not applicable.

3/4.4.1.6 (STS):

See the justification provided for 3/4.4.1.5, above.

3.4.2.1:

The revision to the action statement was made on the basis that the cold overpressure protection system (COPS) will provide the necessary relief capacity as well as assurance that Appendix G limits will not be exceeded. See paragraphs 5.2.2.10 and 5.2.2.11 of the VEGP FSAR.

4.4.3.3:

This surveillance was deleted on the basis that no credit was taken for the operation of the pressurizer heaters in the accident analysis. In addition, paragraph 5.4.10.3.1 of the VEGP FSAR states that "analysis demonstrates that the RCS sensible heat capacity is such that adequate subcooling can be

maintained in the RCS for four hours without heat input from the pressurizer heaters."

3.4.4:

Action Statements a and b were revised to specify hot shutdown rather than cold shutdown on the basis that the applicability for this specification is Modes 1, 2, and 3. Therefore, placing the plant in hot shutdown satisfies the requirement to place the plant in a mode to which the specification does not apply.

The outage time in Action Statement b.2 was revised from 1 hour to 4 hours on the basis that the pressurizer code safety valves will serve to relieve pressure in the event that an overpressure transient occurs during the 4-hour period. The operability requirements on the PORVs are based on the use of the PORVs in achieving safety grade cold shutdown. Since the PORVs are not needed for some time during the shutdown process, a 1-hour limit is overly restrictive and does not allow sufficient time to make a containment entry. The 4-hour limit provides some additional flexibility while maintaining the intent of the specification.

4.4.4.1:

Reference to the requirements of Specification 4.0.5 has been deleted on the basis that the provisions of this specification are applicable without reinstatement in individual specifications. The words "of the actuation instrumentation" were deleted as being redundant. The requirement to perform a channel calibration is sufficiently clear.

4.4.4.3:

This surveillance was deleted on the basis that the normal power supply to the PORVs is Class 1E. See paragraphs 5.2.2.10, 5.4.7.2.3.5, and 5.4.13.2 of the VEGP FSAR.

3/4.4.5:

General: The term "plugged" as it refers to steam generator tubes has been replaced by "repaired or plugged." This is to allow either option to be used as determined appropriate. Repairing of steam generator tubes has been previously approved by the NRC as an acceptable method to correct tube thinning or failure.

4.4.5.0:

The requirements of Specification 4.0.5 have been deleted from this surveillance requirement. The provisions of this specification are applicable without reinstatement in individual specifications.

4.4.5.3.b:

The 20-month inspection period has been changed to a 12- to 24-month inspection interval. This will make 4.4.5.3.b consistent with 4.4.5.3.a and will allow a return to 40-month intervals after two consecutive C-1 category inspections.

4.4.5.3.c:

The term "Condition IV" was added to be consistent with Regulatory Guide 1.83.

4.4.5.4.a.6:

The footnote * was deleted on the basis that it is instructional, and, once the proper value is specified, unnecessary.

4.4.5.5.c:

During the time that steam generator tube inspections are performed, the plant will be in a safe shutdown condition. The fact that a Category C-3 condition is identified does not indicate imminent danger to the health and safety of the public since the plant is in a safe shutdown condition. Therefore, a report to the commission pursuant to 10 CFR 50.73 should be sufficient to keep the commission adequately informed of the situation, especially in light of the fact that the Technical Specifications require the report to be submitted to the commission prior to the resumption of plant operation. In addition, the Technical Specifications require the report to contain a "description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence." A report pursuant to 10 CFR 50.73 is more appropriate in meeting this requirement. It would be unnecessarily difficult and counter-productive to try to provide the required information in an immediate notification. In addition, it should be noted that the NRC has approved this application of the LER rule on the Farley Nuclear Plant.

Table 4.4-1:

This table was modified to reflect the plant's intention to complete a preservice inspection and to reflect the four-loop design.

Table 4.4-2:

The term "plugged" as it refers to steam generator tubes has been replaced by "repaired or plugged."

"Prompt notification to the NRC pursuant to Specification 6.9.1" has been replaced by "Notification to NRC pursuant to paragraph 50.73 of 10 CFR Part 50" as set forth in Generic Letter 83-43 and as discussed above under 4.4.5.5.c.

3.4.6.1.b:

This item was revised to reflect the plant design. See paragraph 5.2.5.3.1 of the VEGP FSAR.

3.4.6.1.c:

The words "not taken credit for in 3.4.6.1.a, above" were inserted for clarification.

4.4.6.1.b:

This item was revised to reflect the plant design. See paragraph 5.2.5.3.1 of the VEGP design.

4.4.6.1.c:

The surveillance was modified to reflect the containment air cooler condensate monitoring system. See paragraph 5.2.5.3.1 of the VEGP FSAR.

3.4.6.2:

Item c was revised on the basis that the VEGP design does not include RCS loop isolation valves.

Item e (STS) has been deleted due to the deletion of the CONTROLLED LEAKAGE definition and the addition of Surveillance Requirement 4.5.2.g.3. The surveillance requirement limits the flowrate across the RCP seal water injection throttle valves at a specified pressure drop across the valve. See the justification provided for the deletion of the definition of CONTROLLED LEAKAGE.

3.4.6.2.e, Action c, and Table 3.4-1:

Table 3.4-1 has been revised to list valve size and the maximum allowable leakage based on the criteria of 0.5 gal/min per inch of nominal valve size up to a maximum of 5 gal/min. Reference to Table 3.4-1 in LCO item e and Action Statement c has been provided for the purpose of clarification. Footnote * was added to allow testing of the RCS pressure isolation valves in accordance with the ASME Boiler and Pressure Vessel Code, Section XI, paragraph IWV-3423.

4.4.6.2.1.c (STS):

Item c was deleted to reflect the deletion of the CONTROLLED LEAKAGE definition and the new Surveillance 4.5.2.g.3.

4.4.6.2.2:

These surveillance requirements have been revised based on the NRC staff position as stated in FSAR Question 210.48 and the VEGP response to that question. In the response to Question 210.48 VEGP stated that periodic leak testing of each valve listed in Table 210.48-1 (Table 3.4-1 of the Technical Specifications) will be performed:

1. After each refueling outage;

2. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve:

3. Following valve actuation due to automatic or manual action, or flow through the valve (for systems rated at less than 50 percent of RCS design pressure).

This is also consistent with subsection 3.9.6 of the VEGP FSAR.

3.4.8:

The words "of gross radioactivity" were added so as to clarify that this specification is inclusive of all forms of activity; not just dose equivalent iodines. This is also consistent with the wording found in Table 4.4-4.

Table 4.4-4:

The words "continuous increase or decrease in thermal power that exceeds" were inserted for clarification. The intent of the specification is maintained. See the justification for definition 1.1C for revision to footnotes * and **. The words "isotopic decay" were added for clarification.

4.4.9.1.1:

The surveillance frequency was revised from 30 minutes to 1 hour on the basis that this is consistent with the limiting condition for operation and the intent of the specification is preserved.

3.4.9.2:

Item c was modified to reflect the fact that the pressurizer was designed to withstand 10 inadvertent auxiliary spray actuations with a spray nozzle delta-T of 625°F. The cyclic or transient limits of 10 inadvertent auxiliary spray actuation cycles (Table 5.7-1) with a spray water temperature differential greater than 320°F reflects the design limits or spray temperature between 320 and 625°F. See paragraph 3.9.N.1.1.2.5 of the VEGP FSAR.

4.4.9.2:

See the justification provided for 4.4.9.1.1.

3.4.9.3:

This specification was revised to: 1) incorporate the setpoint curve for the COPS; 2) allow credit for the RHR suction relief valves for overpressure protection, and; 3) allow several options for venting and depressurizing the RCS. The capacity of the RHR suction relief valves are adequate to provide relief protection necessary for the RHR system and the RCS as part of the cold overpressure protection system. See FSAR paragraph 5.4.7.2.4. The options for venting the RCS might include but would not be limited to: 1) opening the RCS vents; 2) removing a pressurizer code safety valve, and; 3) opening a PORV block valve and removing the PORV valve internals.

4.4.10.1:

This statement was inserted to be consistent with Specification 3.4.10. This will provide clarification. The intent remains unchanged.

4.4.10.3:

This surveillance requirement has been added to ensure compliance with the inservice inspection requirements for piping segments defined as "no break zones" in the high energy line break analysis as discussed in Branch Technical Position MEB 3-1, and VEGP FSAR 3.6.2 and 6.6.

3/4.11:

The specification was modified to reflect the VEGP reactor coolant system vent design. See subsection 5.4.15 of the VEGP FSAR.

Accum	1	2	3	4
Level L/A	950 951	952 953	954 955	956 957

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

Accum	1	2	3	4
Press P/A	960 961	962 963	964 965	966 967

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between ~~[6100]~~ 6616 (36.1% of span) and ~~[6500]~~ 6854 (63.9% of span) gallons,

c. A boron concentration of between [1900] and [2100] ppm, and

d. A nitrogen cover-pressure of between ~~[603]~~ 564 and ~~[666]~~ 637 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying ~~by the absence of alarms,~~ the contained borated water volume and nitrogen cover-pressure in the tanks, and

2) Verifying that each accumulator isolation valve is open. (HV-8808 A,B,C,D)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to [1% of tank volume] 67 gallons by verifying the boron concentration of the accumulator solution; and

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

the circuit breaker supplying

c. At least once per 31 days when the RCS pressure is above [1000 psig] by verifying that power to the isolation valve operator is ~~discon-nected by removal of the breaker from the circuit~~. open.

~~[d. At least once per 18 months by verifying that each accumulator isola-tion valve opens automatically under each of the following conditions:~~

~~1) When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) Setpoint, and~~

~~2) Upon receipt of a Safety Injection test signal.]~~

4.5.1.2 Each accumulator water level and pressure channel shall be demon-strated OPERABLE: by

~~a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and~~

~~b.~~ At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Safety Injection pump ~~(four loop plants only)~~,

c. One OPERABLE RHR heat exchanger,

d. One OPERABLE RHR pump, and

e. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and semi-automatically transferring suction to the containment emergency sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

* The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into Mode 3 for the Safety Injection pumps declared inoperable pursuant to Specification 3.5.3.2 provided the Safety Injection pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding 375°F, whichever occurs first.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
HV-8835	SI Pump Cold Leg Inj.	OPEN
HV-8840	RHR Pump Hot Leg Inj.	CLOSED
HV-8813	SI Pump Mini-flow Isol.	OPEN
HV-8806	SI Pump Suction From RWST	OPEN
HV-8802A,B	SI Pump Hot Leg Inj.	CLOSED
HV-8809A,B	RHR Pump Cold Leg Inj.	OPEN

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping high points, and

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. By a visual inspection which verifies that no loose debris (rags, trash, clothing, etc.) is present in the containment which could be transported to the Containment Emergency Sump and cause restriction of the pump suctions during LOCA conditions. This visual inspection shall be performed:

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System by ensuring that:

a) With a simulated or actual Reactor Coolant System pressure signal greater than or equal to 425 psig the interlocks prevent the valves from being opened, and

b) With a simulated or actual Reactor Coolant System pressure signal less than or equal to {750} psig the interlocks will cause the valves to automatically close.

2) A visual inspection of the Containment Emergency Sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a (Safety Injection ~~actuation~~ and, for the ~~Automatic Switchover to Containment Sump) test signals, and~~ test signal RHR semi-automatic switchover to Containment Emergency Sump, on RWST level low-low test signal and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

   a) Centrifugal charging pump,

   b) Safety Injection pump, and

   c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1) Centrifugal charging pump ≥ 2364 psid,

2) Safety Injection pump ≥ 1405 psid, and

3) RHR pump ≥ 190 psid.

g. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE, and

2) At least once per 18 months.

HPSI System Valve Number		~~LPSI System Valve Number~~
1204-U4-022	1204-U4-139	
1204-U4-023	1204-U4-140	______
1204-U4-024	1204-U4-141	______
1204-U4-025	1204-U4-142	______
1204-U4-116	1204-U4-414	______
1204-U4-117	1204-U4-415	
1204-U4-118	1204-U4-416	
1204-U4-119	1204-U4-417	

See insert for 3/4 5-5.

Insert for Page 3/4 5-5

3) Within 4 hours following completion of each valve manipulation or maintenance of the RCP Seal Injection throttle valves listed in 4.5.2.g.2, by performing a verification test that ensures the seal injection flow is throttled to give a pressure differential of 100 psid ± 15 psid at 32 gpm as measured between the charging pump discharge header and the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 284 gpm, and

b) The total pump flow rate is less than or equal to 550 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 406 gpm, and

b) The total pump flow rate is less than or equal to 650 gpm.

3) For RHR pump lines, with a single pump running, the sum of the injection line flow rates is greater than or equal to 2791 gpm.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

ECCS SUBSYSTEM
LIMITING CONDITION FOR OPERATION

3.5.3.1 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of emergency operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

~~*A maximum of one centrifugal charging pump and one Safety Injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F.~~

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

~~4.5.3.2 All charging pumps and Safety Injection pumps, except the above allowed OPERABLE pumps, shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to [275]°F.~~

3/4.5.3 ECCS SUBSYSTEMS - $T_{avg} \leq 350^{\circ}F$

SAFETY INJECTION PUMPS

LIMITING CONDITION FOR OPERATION

3.5.3.2 All Safety Injection pumps shall be inoperable.

APPLICABILITY: Modes 4, 5, and 6 with the reactor vessel head on.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to an inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS

4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below 325°F, and at least once per 31 days thereafter.

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* An inoperable pump may be energized for testing or for filling accumulators provided the discharge at the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK [OPTIONAL IF ~~NOT~~ TAKEN CREDIT FOR IN PLANT-SPECIFIC ANALYSES]

LIMITING CONDITION FOR OPERATION

3.5.4.1 The boron injection tank shall be OPERABLE with:

a. A contained borated water volume of between ______ and ______ gallons,

b. A boron concentration of between 20,000 and 22,500 ppm, and

c. A minimum solution temperature of 145°F.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days,

b. Verifying the boron concentration of the water in the tank at least once per 7 days, and

c. Verifying the water temperature at least once per 24 hours.

BORON INJECTION SYSTEM

HEAT TRACING [OPTIONAL IF BIT TANK IS ~~NOT~~ TAKEN CREDIT FOR IN PLANT-SPECIFIC ANALYSES]

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least two independent channels of heat tracing shall be OPERABLE for the boron injection tank and for the heat traced portions of the associated flow paths.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one channel of heat tracing on either the boron injection tank or on the heat traced portion of an associated flow path OPERABLE, operation may continue for up to 30 days provided the tank and flow path temperatures are verified to be greater than or equal to [145]°F at least once per 12 hours; otherwise, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.4.2 Each heat tracing channel for the boron injection tank and associated flow path shall be demonstrated OPERABLE:

a. At least once per 31 days by energizing each heat tracing channel, and

b. At least once per 24 hours by verifying the tank and flow path temperatures to be greater than or equal to [145]°F. The tank temperature shall be determined by measurement. The flow path temperature shall be determined by either measurement or recirculation flow until establishment of equilibrium temperatures within the tank.

(559.8 inches from tank bottom, 78 % of instrument span)

3/4.5.~~5~~ 4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.~~5~~ 4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of ~~between~~ 631,478 and gallons,

b. A boron concentration of between [2000] ppm and ~~[2200]~~ 2100 ppm of boron,

c. A minimum solution temperature of ~~[35]~~ 50°F, and

d. A maximum solution temperature of ~~[100]~~ 120°F

e. RWST Sludge Mixing Pump Isolation valves capable of closing on RWST low-level.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the RWST inoperable, except for the Sludge Mixing Pump Isolation Valves restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With a Sludge Mixing Pump Isolation Valve(s) inoperable, restore the valve(s) to OPERABLE status within 24 hours or isolate the Sludge Mixing System by either closing the manual isolation valves or deenergizing the solenoid pilot valve to the Sludge Mixing Pump Isolation Valve(s) within 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.~~5~~ 4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the [outside] air temperature is less than ~~[35]~~ 50°F. ~~or greater than [100]°F~~.

c. At least once per 18 months by verifying that the sludge mixing pump isolation valves automatically close upon an RWST low-level test signal.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.5

3/4.5:

Instrument tag numbers were added, as appropriate, to assist the operator in ensuring compliance with the Technical Specifications.

4.5.1.1:

Surveillance 4.5.1.1.a.1 was modified to allow the plant to take credit for alternate means of monitoring the accumulator pressure and water volume.

Surveillance 4.5.1.1.c was revised on the basis that placing the motor control center circuit breaker to the open or off position will remove power from the valve and preclude inadvertent operation.

4.5.1.1.d:

Since the Technical Specifications require that power be removed from the accumulator isolation valves at all times during Modes 1 and 2 and in Mode 3 when pressurizer pressure is above 1000 psig, and that the valves be verified to be open at least once per 12 hours, the above requirement is unnecessary. Since the valves are required to be deenergized, the question of whether or not the valves open on an actuation signal is of academic importance only.

This deviation from the STS has already been approved and implemented in the Callaway Unit 1 Technical Specifications.

4.5.1.2:

4.5.1.2.a was deleted to reflect the fact that the accumulator is not demonstrated operable (4.5.1.1.a.1) strictly by the absence of alarms, but can be demonstrated operable by meter or the process computer.

3.5.2:

The transfer of suction from the RWST to the containment emergency sump is a semiautomatic operation in that the transfer is automatically initiated, but operator action is required to complete the transfer. See paragraph 6.3.2.8 of the VEGP FSAR. Footnote * was added to allow Mode changes consistent with the operational requirements of the COPS.

4.5.2.c & 4.5.2.d.2:

The ECCS takes suction from the containment emergency sump in the recirculation mode of operation. See paragraph 6.3.2.8 of the VEGP FSAR.

4.5.2.e:

This surveillance was revised to reflect the semiautomatic nature of the switchover from the RWST to the containment emergency sump. See paragraph 6.3.2.8 of the VEGP FSAR.

4.5.2.g.3:

The generic STS surveillance (4.4.6.2.1.c) ensures that the flow through the seal water injection line is less than that assumed in the accident analysis by measuring controlled leakage at a specified RCS pressure with the modulating flow control valve (121) fully open.

This method is sensitive to conditions in the chemical and volume control system, particularly the position of the charging flow valve (182). The proposed Technical Specification change would provide a more quantitative measurement of the seal water line flow by limiting the flowrate across the RCP seal water injection throttle valves at a specified pressure drop across the valve. This will result in making the measurement independent of charging flow valve (182) position, modulating flow control valve (121) position, or CVCS system conditions.

3/4 5.3.1:

The ECCS takes suction from the containment emergency sump in the recirculation mode of operation. See paragraph 6.3.2.8 of the VEGP FSAR.

In addition, footnote * was deleted to reflect the 350°F arming temperature for the COPS which covers all of Mode 4. All charging pumps are allowed to be operable below 350°F. Surveillance Requirement 4.5.3.2 (STS) was deleted on the basis that a separate specification has been created which requires the safety injection pumps to be inoperable below 350°F and the charging pumps are allowed to be operable below 350°F.

See paragraph 5.2.2.10 of the VEGP FSAR.

3/4.5.3.2:

This specification was added to ensure inoperability of the safety injection pumps. The cold overpressurization analysis assumes mass injection from all three charging pumps. The Technical Specifications are written to allow the charging pumps to be operable and to allow mode changes. This specification was approved for Callaway Unit 1. See paragraph 5.2.2.10 of the VEGP FSAR.

3.5.4.1 BORON INJECTION SYSTEM (STS):

Deleted to reflect plant accident analysis assumptions. See paragraph 6.3.2.2.2 of the VEGP FSAR.

3.5.4.2 HEAT TRACING (STS):

Deleted to reflect plant accident analysis assumptions. See paragraph 6.3.2.2.2 of the VEGP FSAR.

3.5.4:

An upper limit on the RWST water volume is not specified on the basis that the maximum amount of water that can be transferred to the containment sump without exceeding an acceptable pH exceeds the tank capacity. Therefore, an upper limit is not necessary.

The words "the next" were added to the action statement to be consistent with Action b to Specification 3.1.2.6.

4.5.4.b:

See the justification provided for 4.1.2.6.b.

4.5.4.c:

This surveillance requirement was added to be consistent with paragraph 7.6.5.2 of the VEGP FSAR and paragraph 7.6.2.3 of the VEGP FSAR.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions~~, except as provided in Table 3.6-1 of Specification 3.6.4.1~~;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and

c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a, ~~[30~~ 45 psig~~]~~, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than 0.60 L_a.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days. The blind flange on the fuel transfer canal need not be verified closed except after each draining of the canal.

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of ~~:~~

~~1)~~ Less than or equal to L_a, ~~[~~0.20~~]~~% by weight of the containment air per 24 hours at a pressure not less than P_a, ~~[50~~ 45 psig~~]~~, or

~~2) Less than or equal to L_t, [0.14]% by weight of the containment air per 24 hours at a reduced pressure of P_t, [25 psig].~~

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to a pressure not less than P_a.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With either the measured overall integrated containment leakage rate exceeding 0.75 L_a ~~or 0.75 L_t, as applicable~~, or the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, restore the overall integrated leakage rate to less than 0.75 L_a ~~or less than 0.75 L_t, as applicable~~, and the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-[1972]:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than ~~either~~ P_a, [~~50~~ 45 psig], ~~or at P_t, [25 psig]~~ 21, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet ~~either~~ 0.75 L_a ~~or 0.75 L_t~~, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet ~~either~~ 0.75 L_a ~~or 0.75 L_t~~, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet ~~either~~ 0.75 L_a ~~or 0.75 L_t~~ at which time the above test schedule may be resumed;

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1) Confirms the accuracy of the test by verifying that the absolute value of the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, is equal to or less than 0.25 L_a ~~or 0.25 L_t~~;

2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and

3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between 0.75 L_a and 1.25 L_a. ~~or 0.75 L_t and 1.25 L_t~~.

d. Type B and C tests shall be conducted with gas at a pressure not less than P_a, ~~[50 psig]~~ 45 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks,

2) Purge supply and exhaust isolation valves with resilient material seals,

~~3) Penetrations using continuous Leakage Monitoring Systems, and~~

~~4) Valves pressurized with fluid from a Seal System~~.

e. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;

f. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.~~8.3~~ 7.2 or 4.6.1.~~8.4~~ 7.3, as applicable;

~~g. Type B periodic tests are not required for penetrations continuously monitored by the Containment Isolation Valve and Channel Weld Pressurization Systems provided the systems are OPERABLE by the requirements of Specification 4.6.1.4~~;

SURVEILLANCE REQUIREMENTS (Continued)

~~h.~~ ~~Leakage from isolation valves that are sealed with fluid from a Seal System may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the Seal System and valves are pressurized to at least 1.10 P_a, [55 psig], and the seal system capacity is adequate to maintain system pressure for at least 30 days~~;

~~i.~~ ~~Type B tests for penetrations employing a continuous Leakage Monitoring System shall be conducted at P_a, [50 psig], at intervals no greater than once per 3 years; and~~

g. ~~j.~~ The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, [~~50~~] psig.
45

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage is less than [0.01] L_a as determined by precision flow measurements when measured for at least [30] seconds with the volume between the seals at a constant pressure of [~~50~~ 45 psig];

b. By conducting overall air lock leakage tests at not less than P_a, [~~50~~ 45 psig], and verifying the overall air lock leakage rate is within its limit:

   1) At least once per 6 months,* and

   2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

---

*The provisions of Specification 4.0.2 are not applicable.

**This represents an exemption to Appendix J, paragraph III.D.2 of 10 CFR Part 50. ~~[Applicant must request this exemption.]~~

CONTAINMENT SYSTEMS

CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.1.4 The Containment Isolation Valve and Channel Weld Pressurization Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Containment Isolation Valve or Channel Weld Pressurization System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The Containment Isolation Valve Pressurization System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a, [55 psig], and has adequate capacity to maintain system pressure for at least 30 days.

4.6.1.4.2 The Containment Channel Weld Pressurization System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to P_a, [50 psig], and has adequate capacity to maintain system pressure for at least 30 days.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~3~~4 Primary containment internal pressure shall be maintained between -1.5 and +3.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~3~~4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~8~~ 5 Primary containment average air temperature shall not exceed 120 °F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the containment average air temperature greater than 120 °F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~8~~ 5 The primary containment average air temperature shall be the arithmetical average of the temperatures at the following locations and shall be determined at least once per 24 hours:

Location	Tag Numbers*
a. Level 2	TE-2563
b. Level B	TE-2613
c. Level C	TE-2612
d.	
e.	

* Or local sample at corresponding locations.

CONTAINMENT STRUCTURAL INTEGRITY ~~[Prestressed concrete containment with ungrouted tendons and hemispherical dome.]~~

LIMITING CONDITION FOR OPERATION

3.6.1.~~7~~ 6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.~~7~~ 6.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.7, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~7~~ 6.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment vessel structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least ~~11~~ 13 tendons (4 inverted U and ~~7~~ 9 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. Unless there is abnormal degradation of the containment during the first three inspections, the sample population for subsequent inspections shall include at least ~~6~~ 5 tendons (~~3~~ 2 inverted U and 3 hoop). This test shall include essentially complete detensioning of each tendon to determine if any strands are broken or damaged.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY [Prestressed concrete containment with ungrouted tendons and typical dome.]

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a. at a level below the acceptance criteria of Specification 4.6.1.7, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 15 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated at the end of 1, 3, and 5 years following the initial containment structural integrity test and at 5-year intervals thereafter. The tendons' structural integrity shall be demonstrated by:

a. Determining that a random but representative sample of at least 19 tendons (5 dome, 6 vertical, and 8 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. Unless there is abnormal degradation of the containment during the first three inspections, the sample population for subsequent inspections shall include at least 10 tendons (3 dome, 3 vertical, and 4 hoop);

SURVEILLANCE REQUIREMENTS (Continued)

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (dome, vertical, and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

   1) The tendon wires or strands are free of corrosion, cracks, and damage,

   2) There are not changes in the presence or physical appearance of the sheathing filler-grease, and

   3) A minimum tensile strength of [240,000] psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and

Dome	[143] ksi
Vertical	[147] ksi
Hoop	[140] ksi

e. Verifying the OPERABILITY of the sheathing filler grease by:

   1) No voids in excess of 5% of the net duct volume,

   2) Minimum grease coverage exists for the different parts of the anchorage system, and

   3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.7.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.7.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.7.3 Containment Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

SURVEILLANCE REQUIREMENTS (Continued)

Replace with insert to page 3/4 6-10.

b. ~~Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (inverted U and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:~~

1) The tendon wires or strands are free of corrosion, cracks, and damage,

2) There are not changes in the presence or physical appearance of the sheathing filler-grease, and

3) A minimum tensile strength of ~~[240,000]~~ psi (guaranteed ultimate strength of the tendon material) for at least three ~~wire or~~ strand samples (one from each end and one at mid-length) cut from each removed ~~wire or~~ strand. Failure of any one of the ~~wire or~~ strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6%. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages;

d. Assuring the observed lift-off forces are between the maximum and minimum ~~stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and~~ values given in the tendon surveillance procedure.

~~Inverted U~~	~~[139] ksi~~
~~Hoop: Cylinder~~	~~[147] ksi~~
~~Dome~~	~~[134] ksi~~

e. Verifying the OPERABILITY of the sheathing filler grease by:

1) No voids in excess of 5% of the net duct volume,

2) Minimum grease coverage exists for the different parts of the anchorage system, and

3) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

Insert for page 3/4 6-10

Removing a strand from one hoop tendon and one inverted U tendon and performing an inspection and material test. It shall be determined that over the entire length of both removed strands:

SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.~~7~~6.2 End Anchorages and Adjacent Concrete Surfaces. The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.~~7~~.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.~~7~~6.3 Containment Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY [Reinforced concrete containment]

LIMITING CONDITION FOR OPERATION

3.6.1.7 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.7.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Containment Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.7.2 Reports. Any abnormal degradation of the containment structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.~~8~~ 7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 24 ~~[42-inch]~~ containment shutdown purge supply and exhaust isolation valve shall be closed and sealed closed, and

b. The 14 ~~[8-inch]*~~ containment purge supply and exhaust isolation valve(s) may be open. ~~for up to [1000]* hours during a calendar year provided no more than one pair (one supply and one exhaust) are open at one time.~~

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With a 24 ~~[42-inch]~~ containment purge supply and/or exhaust isolation valve open or not sealed closed, close and seal that valve or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With the [8-inch]* containment purge supply and/or exhaust isolation valve(s) open for more than [1000]* hours during a calendar year, close the open [8-inch]* valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.~~

b. ~~c.~~ With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specifications 4.6.1.~~8.3~~ 7.2 and/or 4.6.1.~~8.4~~ 7.3, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.~~8~~ 7.1 Each 24 ~~[42-inch]~~ containment purge supply and exhaust isolation valve shall be verified to be sealed closed at least once per 31 days.

~~4.6.1.8.2 The cumulative time that all (8-inch) purge supply and exhaust isolation valves have been open during a calendar year shall be determined at least once per 7 days.~~

~~*For a 3-inch valve or less, the valves may be open continuously. For an 8-inch valve or less, the valves may be open for up to 1000 hours during a calendar year. For an 18-inch valve or less, the valves may be open for up to 500 hours during a calendar year. For a valve greater than 18 inches, the valve may be open for up to 250 hours during a calendar year. All valves that may be open during plant operations (MODE 1, 2, 3, or 4) must be qualified to close under postulated accident conditions. After operational experience, the licensee may request additional time for the qualified valves to be open during plant operations and shall provide justification for the requested additional time as indicated in the Bases.~~

SURVEILLANCE REQUIREMENTS (Continued)

7.2 24

4.6.1.~~8.3~~ At least once per 6 months on a STAGGERED TEST BASIS, the inboard and outboard isolation valves with resilient material seals in each sealed closed ~~[42-inch]~~ containment purge supply and exhaust penetration shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than ~~[0.05]~~ L_a when pressurized to P_a.

7.3 14

4.6.1.~~8.4~~ At least once per 3 months each ~~[8-inch]~~ containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than ~~[0.01]~~ L_a when pressurized to P_a.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM ~~[Credit taken for iodine removal]~~

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a differential ~~discharge~~ pressure of greater than or equal to 196 psi~~g~~d when tested pursuant to Specification 4.0.5;

c. At least once per 18 months during shutdown, by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment spray actuation test signal, and

   2) Verifying that each spray pump starts automatically on a Containment spray actuation test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM ~~[OPTIONAL]~~

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with: [L/A-931A,B]

a. A spray additive tank containing a volume of between 3700(87.7%) and 4000(97.4%) gallons of between 30 and 32 % by weight NaOH solution, and

b. Two spray additive eductors each capable of adding NaOH solution from the ~~chemical~~ spray additive tank to a Containment Spray System pump flow.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. At least once per 6 months by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. At least once per 18 months during shutdown, by verifying that each automatic valve in the flow path actuates to its correct position on a Containment spray actuation test signal; and

d. At least once per 5 years by verifying each solution flow rate (to be determined during preoperational tests) from the following drain connections in the Spray Additive System:

1) 1206-X4-012 ~~[Drain line location]~~ (Later) ± (Later) gpm, and

2) 1206-X4-013 ~~[Drain line location]~~ (Later) ± (Later) gpm.

CONTAINMENT COOLING SYSTEM ~~[OPTIONAL] [Credit taken for iodine removal by spray systems]~~

LIMITING CONDITION FOR OPERATION

3.6.2.3 [Two] independent groups of containment cooling fans shall be OPERABLE with ~~[two]~~ four fans ~~systems~~ to each group. [Equivalent to 100% cooling capacity.]

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1) Starting each fan ~~group~~ from the control room, and verifying that each fan ~~group~~ operates at low speed for at least 15 minutes, and

   2) Verifying a cooling water flow rate of greater than or equal to 1359 gpm ~~to each cooler.~~ per pair of containment fan coolers. (FR1918 + FR1919)

b. At least once per 18 months by verifying that each fan group starts automatically and operates at low speed on a safety injection test signal.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM [No credit taken for iodine removal]

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST and transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one Containment Spray System inoperable and at least [four] containment cooling fans OPERABLE, restore the inoperable Spray System to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two Containment Spray Systems inoperable and at least [four] containment cooling fans OPERABLE, restore at least one Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one Containment Spray System inoperable and one group of required containment cooling fans inoperable, restore either the inoperable Spray System or the inoperable group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both the inoperable Spray System and the inoperable group of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to ___ psig when tested pursuant to Specification 4.0.5;

c. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a _______ test signal, and

2) Verifying that each spray pump starts automatically on a _____ test signal.

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

CONTAINMENT COOLING SYSTEM [OPTIONAL] [No credit taken for iodine removal by spray systems]

LIMITING CONDITION FOR OPERATION

3.6.2.3 [Two] independent groups of containment cooling fans shall be OPERABLE with [two] fan systems to each group. [Equivalent to 100% cooling capacity.]

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one group of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore the inoperable group of cooling fans to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling fans inoperable and both Containment Spray Systems OPERABLE, restore at least one group of cooling fans to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling fans to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling fans inoperable and one Containment Spray System inoperable, restore either the inoperable group of containment cooling fans or the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both the inoperable group of containment cooling fans and the inoperable Spray System to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1) Starting each fan group from the control room and verifying that each fan group operates for at least 15 minutes, and

2) Verifying a cooling water flow rate of greater than or equal to ___ gpm to each cooler.

b. At least once per 18 months by verifying that each fan group starts automatically on a ______ test signal.

3/4.6.3 IODINE CLEANUP SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent Iodine Cleanup Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Iodine Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each Iodine Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedures guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is _________ cfm ± 10%.

   2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

   3) Verifying a system flow rate of _____ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ______ cfm ± 10%;

   2) Verifying that the system starts on either a Safety Injection test signal or on a Containment Pressure-High test signal;

   3) Verifying that the filter cooling bypass valves can be opened by operator action; and

   4) Verifying that the heaters dissipate _______ ± _______ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and by leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _______ cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _____ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation).

**Value applicable will be determined by the following equation:
$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

3/4.6.~~4~~3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.~~4~~3 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.~~4~~3.1 The isolation valves specified in Table 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.~~4~~3.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

~~b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and~~

b. ~~c.~~ Verifying that on a Containment ~~Purge and Exhaust~~ Ventilation Isolation test signal, each purge supply and exhaust valve actuates to its isolation position.

4.6.~~4~~3.3 The isolation time of each power-operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Phase "A" Isolation		
a.		
b.		
2. Phase "B" Isolation		
a.		
b.		
3. Containment Purge and Exhaust		
a.		
b.		
4. Manual		
a.		
b.		
5. Other		
a.		
b.		

*May be opened on an intermittent basis under administrative control.
**Not subject to Type C leakage tests.
***The provisions of Specification 3.0.4 are not applicable.

TABLE 3.6-1 (SHEET 1 OF 3)

CONTAINMENT ISOLATION VALVES

Valve Number	Function	Valve Closure Time (s)
1. Containment Isolation Phase "A" (CIA).		
HV-3502	Hot leg sample line and gross failed fuel detector	≤15
HV-3548	Hot leg sample line and gross failed fuel decector	≤15
HV-8220	Hot leg sample line and gross failed fuel detector	≤15
HV-8823	Safety injection to cold leg	≤15
HV-8824	Safety injection to hot leg	≤15
HV-8843	Boron injection line to cold leg	≤15
HV-8881	Safety injection to hot leg	≤15
HV-27091	Fire protection water	≤20
HV-8871	Accumulator test and drain line	≤15
HV-8964	Accumulator test and drain line	≤15
HV-8888	Accumulator test and drain line	≤15
HV-8880	Nitrogen supply to accumulator	≤15
HV-8160	Normal letdown line	≤15
HV-8152	Normal letdown line	≤15
HV-8100	Excess letdown and seal water leakoff	≤15
HV-8112	Excess letdown and seal water leakoff	≤15
HV-8825	RHR pump discharge to hot leg	≤15
HV-8890A	RHR loop into cold leg	≤15
HV-8890B	RHR loop into cold leg	≤15
HV-8033	Pressurizer relief tank sample to waste gas compressor suction	≤15
HV-8047	Pressurizer relief tank sample to waste gas compressor suction	≤15

Valve Number	Function	Valve Closure Time (s)
HV-8028	Pressurizer relief tank makeup water supply	≤15
HV-3514	Pressurizer steam sample line	≤15
HV-3513	Pressurizer steam sample line	≤15
HV-3507	Pressurizer liquid sample line	≤15
HV-3508	Pressurizer liquid sample line	≤15
HV-10950	Accumulator sample line	≤15
HV-10952	Accumulator sample line	≤15
HV-10951	Accumulator sample line	≤15
HV-10593	Accumulator sample line	≤15
HV-7699	Reactor coolant drain tank pump discharge	≤15
HV-7136	Reactor coolant drain tank pump discharge	≤15
HV-780	Normal containment sump pumps discharge	≤15
HV-781	Normal containment sump pumps dischange	≤15
HV-7126	Reactor coolant drain tank vent and H_2 supply	≤15
HV-7150	Reactor coolant drain tank vent and H_2 supply	≤15
HV-9385	Service air and post-LOCA purge air supply	≤20
HV-9378	Instrument air	≤15
HV-8211	Post-accident sampling	≤15
HV-8212	Post-accident sampling	≤15
HV-8208	Post-accident sampling	≤15
HV-8209	Post-accident sampling	≤15
2. Containment Ventilation Isolation (CVI).		
HV-12975	Containment air radioactivity monitor inlet	≤15
HV-12976	Containment air radioactivity monitor inlet	≤15

TABLE 3.6-1 (SHEET 3 of 3)

Valve Number	Function	Valve Closure Time (s)
HV-12977	Containment air radioactivity monitor outlet	≤ 15
HV-12978	Containment air radioactivity monitor outlet	≤ 15
HV-2626A	Containment pre-access purge supply and equalizing	≤ 10
HV-2627A	Containment pre-access purge supply and equalizing	≤ 10
HV-2626B	Containment mini-purge supply and equalizing	≤ 5
HV-2627B	Containment mini-purge supply and equalizing	≤ 5
HV-2628A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2629A	Containment pre-access purge exhaust and equalizing	≤ 10
HV-2628B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2629B	Containment mini-purge exhaust and equalizing	≤ 5
HV-2624A	Post-accident air exhaust	≤ 20
HV-2624B	Post-accident air exhaust	≤ 20

CONTAINMENT SYSTEMS

3/4.6.~~3~~4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.~~3~~4.1 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one hydrogen monitor inoperable, restore the inoperable monitor to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.~~3~~4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, an ANALOG CHANNEL OPERATIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

~~a. One volume percent hydrogen, balance nitrogen, and~~

~~b.~~ Four volume percent hydrogen, balance nitrogen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

3.6.~~5~~4.2 Two independent Hydrogen Recombiner Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Recombiner System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.~~5~~4.2 Each Hydrogen Recombiner System shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a Hydrogen Recombiner System functional test, that the minimum heater sheath temperature increases to greater than or equal to [700]°F within [90] minutes. Upon reaching [700]°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to [60] kW, and

b. At least once per 18 months by:

    1) Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

    2) Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

    3) Verifying the integrity of all heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

CONTAINMENT SYSTEMS

HYDROGEN PURGE CLEANUP SYSTEM [If less than 2 hydrogen recombiners available]

LIMITING CONDITION FOR OPERATION

3.6.5.3 A Hydrogen Purge Cleanup System shall be OPERABLE and capable of being powered from a minimum of one OPERABLE emergency bus.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the Hydrogen Purge Cleanup System inoperable, restore the Hydrogen Purge Cleanup System to OPERABLE status within 30 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.3 The Hydrogen Purge Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ______ cfm ± 10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%: and

   3) Verifying a system flow rate of ______ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

   1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of ___ cfm ± 10%,

   2) Verifying that the filter cooling bypass valves can be manually opened, and

   3) Verifying that the heaters dissipate _____ ± _____ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%; and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of ______ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

HYDROGEN MIXING SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.5.4 Two independent Hydrogen Mixing Systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one Hydrogen Mixing System inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.5.4 Each Hydrogen Mixing System shall be demonstrated OPERABLE:

a. At least once per 92 days on a STAGGERED TEST BASIS by starting each system from the control room and verifying that the system operates for at least 15 minutes, and

b. At least once per 18 months by verifying a system flow rate of at least _______ cfm.

CONTAINMENT SYSTEMS

3/4.6.6 PENETRATION ROOM EXHAUST AIR CLEANUP SYSTEM [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.6 Two independent Penetration Room Exhaust Air Cleanup Systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Penetration Room Exhaust Air Cleanup System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 Each Penetration Room Exhaust Air Cleanup System shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

1) Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is _______ cfm ± 10%;

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.5.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%; and

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying a system flow rate of ___ cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation, by verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than [**]%;

d. At least once per 18 months by:

1) Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than [6] inches Water Gauge while operating the system at a flow rate of _____ cfm ± 10%,

2) Verifying that the system starts on a Safety Injection test signal,

3) Verifying that the filter cooling bypass valves can be manually opened, and

4) Verifying that the heaters dissipate _____ ± _______ kW when tested in accordance with ANSI N510-1975.

e. After each complete or partial replacement of a HEPA filter bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a DOP test aerosol while operating the system at a flow rate of _____ cfm ± 10%, and

f. After each complete or partial replacement of a charcoal adsorber bank, by verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than [*]% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of _______ cfm ± 10%.

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*0.05% value applicable when a HEPA filter or charcoal adsorber efficiency of 99% is assumed, or 1% when a HEPA filter or charcoal adsorber efficiency of 95% or less is assumed in the NRC staff's safety evaluation. (Use the value assumed for the charcoal adsorber efficiency if the value for the HEPA filter is different from the charcoal adsorber efficiency in the NRC staff's safety evaluation.)

**Value applicable will be determined by the following equation:

$P = \frac{100\%-E}{SF}$, when P equals the value to be used in the test requirement (%), E is efficiency assumed in the SER for methyl iodide removal (%), and SF is the safety factor to account for charcoal degradation between tests (5 for systems with heaters and 7 for systems without heaters).

3/4 6.7 VACUUM RELIEF VALVES [OPTIONAL]

LIMITING CONDITION FOR OPERATION

3.6.7 The primary containment to atmosphere vacuum relief valves shall be OPERABLE with an Actuation Setpoint of less than or equal to ___ psid.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one primary containment to atmosphere vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.7 No additional requirements other than those required by Specification 4.0.5.

JUSTIFICATIONS FOR DEVIATIONS FROM STS

SECTION 3/4.6

3/4.6:

Instrument tag numbers were added as appropriate to assist the operator in ensuring the compliance with these Technical Specifications.

4.6.1.1.a, Footnote *:

The blind flange on the fuel transfer canal is only used during a refueling outage and, therefore, surveillance should not be required to be as frequent. In addition, access is limited both physically and from an ALARA standpoint. See paragraph 9.1.2.2 of the VEGP FSAR.

4.6.1.1.a:

Surveillance 4.6.1.1.a has been revised to delete the reference to Table 3.6-1 on the basis that Table 3.6-1 is a listing of only those valves which receive a containment isolation phase A or a containment ventilation isolation signal. These are the valves which are required to close within the time limits specified so that the assumptions made for the LOCA analysis are verified. Those penetrations covered by 4.6.1.1.a will be administratively controlled and need not be listed in these Technical Specifications. Any penetrations which may have to be opened on an intermittent basis and are required to be closed during accident conditions are capable of being closed by automatic containment isolation valves.

Valves which are required to be open during accident conditions should not be listed in Table 3.6-1 on the basis that they were assumed to be open from the standpoint of the safety analysis. See subsection 6.2.4 of the VEGP FSAR.

3/4.6.1.2:

All references to reduced pressure testing (L_t) have been deleted in accordance with Readiness Review Module 3A.

3.6.1.2.a:

The words "a pressure not less than" were provided to clarify that the test pressure specified is a lower bound. This will ensure that the local/integrated leak testing will be in compliance with these specifications and is therefore a revision in the conservative direction.

3.6.1.2.b:

See the justification provided for 3.6.1.2.a.1.

4.6.1.2.c.1:

The words "the absolute value of" were inserted to provide clarification.

4.6.1.2.d.3 and 4.6.1.2.d.4:

VEGP does not use a penetration continuous leakage monitoring system or take credit for isolation valves sealed with fluid from a seal system.

4.6.1.2.g (STS):

This item was deleted on the basis that the VEGP is not equipped with a containment isolation valve or channel weld pressurization system. See subsection 6.2.1 of the VEGP FSAR.

4.6.1.2.h (STS):

This item was deleted because it does not apply to the VEGP. See subsection 6.2.1 of the VEGP FSAR.

4.6.1.2.i (STS):

This item was deleted because the VEGP is not equipped with a penetration continuous leakage monitoring system. See subsection 6.2.1 of the VEGP FSAR.

3/4.6.1.4 (STS):

This specification was deleted on the basis that the VEGP is not equipped with a containment isolation valve or channel weld pressurization system. See subsections 6.2.1 and 6.2.6 of the VEGP FSAR.

3/4.6.1.7, Prestressed Concrete Containment With Ungrouted Tendons and Typical Dome and Reinforced Concrete Containment (STS):

These specifications were deleted as not applicable. As discussed in paragraph 3.8.1.1 of the VEGP FSAR, the VEGP containment consists of a prestressed reinforced concrete cylinder and hemispherical dome. Therefore, the applicable specification for containment structural integrity is used.

4.6.1.6:

The VEGP commitment for surveillance of the post-tensioning system is to follow Revision 2 of Regulatory Guide 1.35. The surveillance requirements for the containment structural integrity section, as revised, meet this commitment. Since the basis for the Standard Technical Specification is also Regulatory Guide 1.35, the surveillance section, as written, is not a deviation from the Standard Technical Specification, but an expansion of it to include specifics of how surveillance will be conducted for VEGP. See paragraph 3.8.1.7.2 of the VEGP FSAR.

4.6.1.6.1.a and 4.6.1.6.1.b:

These surveillance requirements were revised to reflect the VEGP commitment to Regulatory Guide 1.35, Rev. 2, as discussed in subsection 1.9.35 of the VEGP FSAR. Specifically, the appropriate number of tendons and the requirement to essentially detension all tendons of the sample have been specified.

4.6.1.6.1.d:

The specific lift-off stress acceptance criteria was replaced with reference to values given in the tendon surveillance procedure because the predicted acceptable limits for the lift-off forces for each tendon are a function of time, initial tensioning, and location in the containment structure. The VEGP acceptance criteria are developed consistent with the guidelines of Draft Regulatory Guide 1.35.1.

3/4.6.1.7:

The VEGP minipurge system is designed to reduce airborne radiation due to noble gases in the containment atmosphere prior to an entry and relieve pressure buildup due to excess air leakage into the containment from pneumatic controllers. The system includes a series pair of supply valves and a series pair of exhaust valves. The minipurge system has been designed to operate continuously to minimize personnel exposures during containment entries. The VEGP minipurge system meets the requirements of Branch Technical Position CSB 6-4 as discussed FSAR subsection 9.4.6. The minipurge isolation valves are qualified and are able to operate under DBA conditions. These valves are included in the pump and valve operability program discussed in FSAR subsection 3.9.B.3. The VEGP accident analysis has considered the time it takes to isolate the purge lines post-LOCA and has shown that the 10 CFR 100 dose limits are not exceeded (FSAR subsection 15.6.5). Operation of the minipurge system has also been considered in the ECCS backpressure analysis (FSAR subsection 6.2.1). This analysis demonstrated that there is sufficient backpressure in containment for the ECCS system to function properly, even with the minipurge system operating. Therefore, the continuous operation of the minipurge system will not prevent the safe shutdown of the plant nor will it cause 10 CFR 100 dose limits to be exceeded.

3/4.6.2.1, Containment Spray System (No credit taken for iodine removal):

This specification was deleted in its entirety since the comparable specification with credit taken for iodine removal was utilized. See subsection 6.5.2 of the VEGP FSAR.

4.6.2.1.b:

This surveillance was revised to call for differential pressure as opposed to discharge pressure on the basis that the intent of this specification is to verify that the performance (head versus flow) of the pump has not degraded to a point below which the current analysis is no longer valid. The use of differential pressure allows direct comparison of test data with the pump acceptance curve without correcting for a varying suction pressure.

3.6.2.3:

The VEGP design includes four fans in each independent group. See paragraph 6.2.2.1 of the VEGP FSAR.

4.6.2.3:

Item 4.6.2.3.a.1 and 4.6.2.3.b have been modified to indicate that the containment cooling fans would operate in slow speed following a safety injection and should be demonstrated OPERABLE in slow speed. Item 4.6.2.3 reflects plant design. See paragraph 6.2.2.1 of the VEGP FSAR. Item 4.6.2.3.a.2 was revised to reflect the fact that flow instrumentation is designed to indicate flow to a pair of containment coolers.

3/4.6.2.3, Containment Cooling System (No Credit Taken For Iodine Removal By Spray Systems) (STS):

This specification was deleted in its entirety since the comparable specification with credit taken for iodine removal was utilized. See subsection 6.5.2 of the VEGP FSAR.

3/4.6.3 (STS):

This specification was deleted since the VEGP is not equipped with an iodine cleanup system. See subsection 6.5.2 of the VEGP FSAR.

3/4.6.3:

The exclusion from the provisions of Specification 3.0.4 was added on the basis that if an affected penetration is isolated within 4 hours by a deactivated valve, closed manual valve, or a blind flange, containment integrity is preserved and the plant is not in a degraded condition. In addition, should penetration isolation render a safety-related system or

train inoperable, the appropriate action in accordance with the applicable LCO will have to be taken. Therefore, Specification 3.0.4 should not be applicable.

<u>4.6.3.2</u>:

This surveillance requirement was revised to reflect the fact that the VEGP does not use Phase B isolation valves. See subsection 6.2.4 of the VEGP FSAR. In addition, "Purge and Exhaust" was deleted and replaced with "Ventilation" and the word "Supply" was inserted to reflect plant-specific nomenclature.

<u>4.6.4.1</u>:

The requirement that channel calibration be performed using sample gas containing one volume percent hydrogen was deleted based on the fact that the recombiner system is designed to ensure that the containment atmosphere is maintained at 4-percent by volume hydrogen. Sampling with 4-percent sample gas should be sufficient to verify functional capability of the hydrogen monitors.

<u>3/4.6.5.3 (STS)</u>:

This specification was deleted in its entirety on the basis that the VEGP is equipped with redundant hydrogen recombiners. See subsection 6.2.5 of the VEGP FSAR.

<u>3/4.6.5.4 (STS)</u>:

This specification was deleted in its entirety on the basis that the VEGP is not equipped with a hydrogen mixing system. See subsection 6.2.5 of the VEGP FSAR.

<u>3/4.6.6 (STS)</u>:

The VEGP is equipped with an electrical penetration room filtration and exhaust system and an ECCS piping penetration area filtration and exhaust system. No credit was assumed in the safety analysis for the operation of the electrical penetration room filtration and exhaust system. Therefore, this specification has been deleted. The ECCS piping penetration area filtration and exhaust system is covered by Specification 3/4.7.7. See paragraphs 15.6.5.4.6.1 and 9.4.3.2 of the VEGP FSAR.

<u>3/4.6.7 (STS)</u>:

This specification was deleted in its entirety on the basis that the VEGP is not equipped with vacuum relief valves. See subsection 6.2.1 of the VEGP FSAR.