ATTACHMENT 2

**PROPOSED
TECHNICAL SPECIFICATION
CHANGES**

BASES

ADMINISTRATIVE CONTROLS

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to: L_a, 0.5 percent by weight of the containment air per 24 hours at P_a, 56.5 psig.

b. A combined leakage rate of less than or equal to 0.60 L_a for primary containment penetrations and primary containment isolation valves subject to Type B and C tests when pressurized to P_a in accordance with the Primary Containment Leakage Rate Testing Program described in Specification 6.8.5.g, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested.

c. *Less than or equal to 100 scf per hour for all four main steam lines when tested at 25.0 psig.

d. A combined leakage rate of less than or equal to 5 gpm for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a, 62.2 psig.

e. Less than or equal to 1 gpm times the number of valves per penetration not to exceed 3 gpm per penetration for any line penetrating containment and hydrostatically tested at 1.10 P_a, 62.2 psig.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:
With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or
b. The measured combined leakage rate for primary containment penetrations and primary containment isolation valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested, exceeding 0.60 L_a, or
c. The measured leakage rate exceeding 100 scf per hour for all four main steam lines, or
d. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 5 gpm, or
e. The leakage rate of any hydrostatically tested line penetrating primary containment exceeding 1 gpm per isolation valve times the number of containment isolation valves per penetration or greater than 3 gpm per penetration,

prior to increasing reactor coolant system temperature above 200°F, restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. The combined leakage rate for primary containment penetrations and primary containment isolation valves subject to Type B and C tests in accordance with the Primary Containment Leakage Rate Testing Program, except for main steam line isolation valves* and primary containment isolation valves which are hydrostatically tested, tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than or equal to 100 scf per hour for all four main steam lines, and

d. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 5 gpm, and

e. The leakage rate of any hydrostatically tested line penetrating primary containment to less than 1 gpm per isolation valve times the number of containment isolation valves per penetration or less than 3 gpm per penetration.

SURVEILLANCE REQUIREMENTS

4.6.1.2 Perform required primary containment leakage rate testing in accordance with the Primary Containment Leakage Rate Program described in Specification 6.8.5.g.**

*Exemption to Appendix J of 10 CFR Part 50

**Except for LPCI Loop A and B Injection Isolation valves, which are hydrostatically tested in accordance with Specification 4.4.3.2.2 in lieu of this requirement.

THIS PAGE HAS BEEN INTENTIONALLY DELETED

THIS PAGE HAS BEEN INTENTIONALLY DELETED

THIS PAGE HAS BEEN INTENTIONALLY DELETED

THIS PAGE HAS BEEN INTENTIONALLY DELETED

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. Within 7 days following each closing, except when the air lock is being used for multiple entries, then at least once per 30 days, by verifying seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to P_a, 56.5 psig.

b. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been opened during periods when containment integrity was not required. The demonstration shall verify a seal leakage rate less than or equal to 5 scf per hour when the gap between the door seals is pressurized to P_a, 56.5 psig, unless the air lock is tested pursuant to Specification 4.6.1.3.c.2.

c. By conducting an overall air lock leakage test at P_a, 56.5 psig, and by verifying that the overall air lock leakage rate is within its limit:

1. Prior to initial fuel loading and at 30 months* intervals thereafter,

2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been opened during periods when containment integrity was not required, if maintenance which could affect the leak tight integrity of the doors has been performed since the last successful test pursuant to Specification 4.6.1.3.c.1.

d. 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.

**Except that the inner door need not be opened to verify interlock OPERABILITY when the primary containment is inerted, provided that the inner door interlock is tested within 8 hours after the primary containment has been deinerted.

PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the primary containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With the structural integrity of the primary containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 The structural integrity of the exposed accessible interior and exterior surfaces of the primary containment shall be determined during the shutdown for each Type A containment leakage rate test by a visual inspection of those surfaces. This inspection shall be performed prior to the Type A containment leakage rate test and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.5.2 Reports Any abnormal degradation of the primary containment structure detected during the above required inspections shall be reported in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days. This report shall include a description of the condition of the structure, the inspection procedure, the inspection criteria, and the corrective actions taken.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY 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 limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary 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 of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The maximum allowable leakage rate for the primary containment (L_a) is 0.5 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 56.5 psig.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR Part 50 Appendix J, Option B. The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Rate Testing Program", Revision 0, dated September 1995, and Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J", Revision 0, dated July 26, 1995. NRC Regulatory Guide 1.163, Revision 0 endorses NEI 94-01 which in turn identifies ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements" as an acceptable standard regarding leakage-rate test methods, procedures, and analyses.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE (Continued)

The measured leakage rate acceptance criteria of $\leq 0.60\ L_a$ for the combined Type B and C tests and as-left acceptance criterion of $\leq 0.75\ L_a$ for the Type A test ensures a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Primary containment operability is maintained by limiting leakage to $\leq 1.0\ L_a$.

Individual leakage rates specified for the primary containment air lock are addressed in Specification 3.6.1.3.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 Option B with the exception of exemptions granted for main steam isolation valve leak testing and testing the Low Pressure Coolant Injection Inboard Isolation Valves. The program as defined in Specification 6.8.5.g eliminates the need for the previous exemptions granted concerning analyzing the Type A test data and testing airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and

CONTAINMENT SYSTEMS

BASES

PRIMARY CONTAINMENT AIR LOCKS (Continued)

3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment. In the event of an inoperable door interlock, locking shut the inner door will ensure containment integrity while permitting access to the lock for maintenance and surveillance testing.

Maintaining primary containment air locks OPERABLE requires compliance with the leakage rate test requirements of 10 CFR 50, Appendix J as established in the Primary Containment Leakage Rate Testing Program, which has been established to implement 10 CFR Part 50, Appendix J, Option B. The Primary Containment Leakage Rate Testing Program conforms with NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Rate Testing Program" Revision 0, dated September 1995 and Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50 Appendix J", Revision 0, dated July 26, 1995.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 56.5 psig in the event of a LOCA. A visual inspection in conjunction with the Primary Containment Leakage Testing Program is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of less than 56.5 psig does not exceed the maximum allowable pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are maintained closed during a majority of the plant operating time. Maintaining these valves closed (even though they have been qualified to close against the buildup of pressure in primary containment in the event of DBA/LOCA) reduces the potential for release of excessive quantities of radioactive material.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

Purging or venting through the Standby Gas Treatment System (SGTS) imposes a vulnerability factor on the integrity of the SGTS. Should a LOCA occur while the purge pathway is through the SGTS the associated pressure surge, before the purge valves close, may adversely affect the integrity of the SGTS charcoal filters. Therefore, PURGING or VENTING through the SGTS is limited to 90 hours per 365 days. This time limit is not imposed when venting through the SGTS with the 1-inch valves or when PURGING or VENTING through the Reactor Building Ventilation System with any of the purge valves.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit 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.

The 6, 10, 20, and 24 inch purge valves are generally configured in a three (3) valve arrangement at each of the associated purge penetrations. The valves are leak tested by pressurizing between the three valves and a total leakage is determined as opposed to a single valve leakage. Verifying that the measured leakage rate is less than 0.05 L_a for this multi-valve arrangement is more conservative than a limit of 0.05 L_a for a single valve.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

PROCEDURES AND PROGRAMS (Continued)

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM,

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and,

3) Participation in an Interlaboratory Comparison Program 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.

g. Primary Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(O) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 56.5 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.5% of primary containment air weight per day at P_a.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

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.

6.9.1.2 The startup report shall address each of the tests identified in Subsection 14.1.4.8 of 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

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.1.3.1.4.a	Scram discharge vol. vent and drain valve operability
4.3.1.1, Table 4.3.1.1-1, Item 3	RPS Rx Steam Dome Press High cal.
4.3.1.1, Table 4.3.1.1-1, Item 4	RPS Rx Low Water Level - Level 3 cal
4.3.1.1, Table 4.3.1.1-1, Item 5	RPS MSIV Closure cal
4.3.1.1, Table 4.3.1.1-1, Item 6	RPS Main Steam Line Radiation High cal
4.3.1.1, Table 4.3.1.1-1, Item 7	RPS Drywell Pressure High cal
4.3.1.3[a]	RPS Response Time Test
4.3.2.1, Table 4.3.2.1-1, Item 1.a.1	Pri Cont Isolation Actuation Rx Water Low Level - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.2	Pri Cont Isolation Actuation Rx Water Low Level - Level 2 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.a.3	Pri Cont Isolation Actuation Rx Water Low Level - Level 1 cal
4.3.2.1, Table 4.3.2.1-1, Item 1.b	Pri Cont Isolation Actuation Drywell Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.1	Pri Cont Isolation Actuation Main Steam Line Radiation High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.c.2	Pri Cont Isolation Actuation Main Steam Line Press Low cal
4.3.2.1, Table 4.3.2.1-1, Item 1.d	Pri Cont Isolation Actuation Main Steam Line Tunnel Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.e	Pri Cont Isolation Actuation Condenser Press High cal
4.3.2.1, Table 4.3.2.1-1, Item 1.f	Pri Cont Isolation Actuation Turbine Bldg. Area Temp. High cal
4.3.2.1, Table 4.3.2.1-1, Item 2.e	RWCU Isolation Rx Water Low Level - Level 2 channel cal
4.3.2.1, Table 4.3.2.1-1, Item 2.g	RWCU Manual Initiation channel functional test
4.3.2.1, Table 4.3.2.1-1, Item 3.a.1	RCIC Steam Line Flow High DP channel cal
4.3.2.1, Table 4.3.2.1-1, Item 3.a.2	RCIC Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.1	HPCI Steam Line Flow High DP cal
4.3.2.1, Table 4.3.2.1-1, Item 4.a.2	HPCI Steam Line Flow High Time Delay cal
4.3.2.1, Table 4.3.2.1-1, Item 4.e	HPCI Manual Initiation functional test
4.3.2.1, Table 4.3.2.1-1, Item 5.a	RHR S/D Cooling Rx Water Level Low - Level 3 cal
4.3.2.1, Table 4.3.2.1-1, Item 6.b	Sec. Cont. Isolation - Drywell Press High channel cal
4.3.2.3[a]	Isolation Actuation Inst. System Response Time
4.3.3.1, Table 4.3.3.1-1, Item 1.b	CS Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.b	LPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.f	LPCI Riser Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 2.g	LPCI Recirc. Pump Differential Pressure High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.b	HPCI Drywell Press High Cal
4.3.3.1, Table 4.3.3.1-1, Item 3.f	HPCI Manual Initiation
4.3.3.1, Table 4.3.3.1-1, Item 4.f	ADS RPV Low Level 3 Cal
4.3.3.1, Table 4.3.3.1-1, Item 4.i	ADS Manual Inhibit Functional Test
4.3.4, Table 4.3.4-1, Item 2	RPV Press High Cal (ATWS)
4.3.7.4.1, Table 4.3.7.4.-1, Item 1	RPV Press Cal - Remote Shutdown
4.3.7.5, Table 4.3.7.5-1, Item 1	RPV Press Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 11	SRV Position Indic Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 12	CTMT High Range Rad Monitoring Cal Accident Mon.
4.3.7.5, Table 4.3.7.5-1, Item 2.a	RPV Fuel Zone Level Cal Accident Mon
4.3.7.10.c	Loose Part Detection System Cal
4.3.9.1, Table 4.3.9.1-1, Item a	RPV High Water Level 8 Cal FW/Main Turbine Trip
4.3.9.2	FW/Main Turbine Trip LSFT
4.3.11.1, Table 4.3.11.1-1, Item 7	Alt S/D system Rx Water Level instrument operability
4.3.11.1, Table 4.3.11.1-1, Item 8	Alt S/D system Rx Press instrument operability
4.4.2.1.1	SRV Tail Pipe Pressure Switch Cal
4.4.2.1.2	SRV lift set point test
4.4.2.2.b	SRV Low Low Set Pressure setpoint Cal and LSFT
4.4.3.1.b	Drywell Sump Flow/Lvl Monitoring Cal
4.4.3.2.2.a	RCS Pressure Isol Valve Leak Test
4.5.1.d.2.a	ADS System Functional Test
4.6.1.4.d.3	MSIV LCS Press Inst. Cal and DP Calibration

TABLE 4.0.2-1

SURVEILLANCE TEST INTERVALS EXTENDED TO OCTOBER 5, 1996 Cont'd

SURVEILLANCE REQUIREMENT	DESCRIPTION
4.6.2.1.e	Suppression Chamber operability (visual inspection)
4.6.2.1.h	Suppression Chamber operability DW to torus bypass leak test
4.6.3.4	Instr. Excess Flow Check operability
4.6.3.5.b	TIP Explosive Squib operability test
4.6.4.1.b.2.a	Torus/Drywell vacuum breaker setpoint operability
4.6.4.1.b.2.b	Torus/Drywell vacuum breaker position indication cal
4.6.4.1.b.2.c	Torus/Drywell vacuum breaker switch opening gap
4.6.4.2.b.2.a	RB/Torus Vacuum Breaker operability (setpoint)
4.6.4.2.b.2.b	RB/Torus Vacuum Breaker operability (visual)
4.6.4.2.b.2.c	RB/Torus Vacuum Breaker position indication operability
4.7.11.4	Alternative Shutdown Control Circuit Functional Test
4.8.4.2.a.1.a	Primary Containment 4160 Volt Penetration Protective Relay Cal
4.8.4.2.a.1.b	Primary Containment 4160 Volt Penetration Protective Device Integrated Functional Test

TABLE NOTATIONS

(a) The surveillance interval of channels within the same trip system required to be tested at least once every N times 18 months, where N is the total number of channels in the trip system, may be based upon the performance of the surveillance during the fifth refueling outage.