LIST OF FIGURES (Continued)

OPERABLE - OPERABILITY

1.28 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 CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

Pa

1.31a Pa (psig) is ≥ the calculated peak containment internal pressure related to design basis accidents, and is equal to 38 psig.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test 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, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING 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 required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

## 2.2 LIMITING SAFETY SYSTEM SETTINGS

### REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, 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.

## TABLE 2.2.1-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

| FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|
| 1. Intermediate Range Monitor, Neutron Flux - High | ≤ 120/125 divisions of full scale | ≤ 122/125 divisions of full scale |
| 2. Average Power Range Monitor: | | |
| a. Neutron Flux-High, Setdown | ≤ 15% of RATED THERMAL POWER | ≤ 20% of RATED THERMAL POWER |
| b. Flow Biased Simulated Thermal Power - High | | |
| 1) Flow Biased | ≤ 0.58W + 59%, with a maximum of | ≤ 0.58W + 62%, with a maximum of |
| 2) High Flow Clamped | ≤ 113.5% of RATED THERMAL POWER | ≤ 114.9% of RATED THERMAL POWER |
| c. Fixed Neutron Flux - High | ≤ 118% of RATED THERMAL POWER | ≤ 120% of RATED THERMAL POWER |
| d. Inoperative | N.A. | N.A. |
| 3. Reactor Vessel Steam Dome Pressure - High | ≤ 1060 psig | ≤ 1074 psig |
| 4. Reactor Vessel Water Level - Low, Level 3 | ≥ 13.0 inches above instrument zero* | ≥ 11.0 inches above instrument zero |
| 5. Main Steam Line Isolation Valve - Closure | ≤ 10.0% closed | ≤ 12.5% closed |
| 6. DELETED | | |

*See Bases Figure B 3/4 3-1.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Lock close or close, as applicable, the affected system isolation valves within 1 hour and declare the affected system inoperable.

TABLE NOTATIONS

*May be bypassed with reactor steam pressure ≤ 1060 psig and all turbine stop valves closed.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the standby gas treatment system.

(c) DELETED

(d) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system outboard isolation valve RWCU-V-4.

(g) Only valves RHR-V-123A and RHR-V-123B in Valve Group 5 are required for primary isolation.

(h) Manual initiation isolates RCIC-V-8 only and only with a coincident reactor vessel level-low, level 3.

(i) Not required for RHR-V-8 when control is transferred to the alternate remote shutdown panel during operational conditions 1, 2 & 3 and the isolation interlocks are bypassed. When RHR-V-8 control is transferred to the remote shutdown panel under operational modes 1, 2, and 3 the associated key lock switch will be locked with the valve in the closed position. Except RHR-V-8 can be returned to, and operated from, the control room, with the interlocks and automatic isolation capability reestablished in operational conditions 2 and 3 when reactor pressure is less than 135 psig.

## TABLE 3.3.2-2

## ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

| TRIP FUNCTION | TRIP SETPOINT | ALLOWABLE VALUE |
|---|---|---|
| 1. PRIMARY CONTAINMENT ISOLATION | | |
| a. Reactor Vessel Water Level | | |
| 1) Low, Level 3 | ≥ 13.0 inches* | ≥ 11.0 inches |
| 2) Low Low, Level 2 | ≥ -50 inches* | ≥ -57 inches |
| b. Drywell Pressure - High | ≤ 1.68 psig | ≤ 1.88 psig |
| c. Main Steam Line | | |
| 1) DELETED | | |
| 2) Pressure - Low | ≥ 831 psig | ≥ 811 psig |
| 3) Flow - High | ≤ 115.6 psid | ≤ 124.6 psid |
| d. Main Steam Line Tunnel Temperature - High | ≤ 164°F | ≤ 170°F |
| e. Main Steam Line Tunnel ΔTemperature - High | ≤ 80°F | ≤ 90°F |
| f. Condenser Vacuum - Low | ≥ 23 inches Hg absolute pressure | ≥ 24.5 inches Hg absolute pressure |
| g. Manual Initiation | N.A. | N.A. |
| 2. SECONDARY CONTAINMENT ISOLATION | | |
| a. Reactor Building Vent Exhaust Plenum Radiation - High | ≤ 13.0 mR/h | ≤ 16.0 mR/h |
| b. Drywell Pressure - High | ≤ 1.68 psig | ≤ 1.88 psig |
| c. Reactor Vessel Water Level - Low Low, Level 2 | ≥ -50 inches* | ≥ -57 inches |
| d. Manual Initiation | N.A. | N.A. |

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve, and

d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary 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 Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10-month intervals during shutdown at $P_a$, 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.

b. If any periodic Type A test fails to meet 0.75 $L_a$, 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 0.75 $L_a$, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 $L_a$, 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 supplemental test result, $L_c$, minus the sum of the Type A and the superimposed leak, $L_o$, are equal to or less than 0.25 $L_a$.

   2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 $L_a$ and 1.25 $L_a$.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at $P_a$*, at intervals no greater than 24 months*** except for tests involving:

   1. Air Locks

   2. Main steam line isolation valves,

   3. Valves pressurized with fluid from a seal system,

   4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

   5. Purge supply and exhaust isolation valves with resilient seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. 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 $P_a$, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.1 and 4.6.1.8.2.

j. The provisions of Specification 4.0.2 are not applicable to 24-month or 40 ± 10-month surveillance intervals.

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*Unless a hydrosatic test is required per Table 3.6.3-1.

***For those tests conducted during refueling outages, the 24-month interval may be exceeded by no more than 3 months.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. The interlock operable and engaged such that both doors cannot be opened simultaneously, and

b. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, and

c. An overall air lock leakage rate of less than or equal to 0.05 $L_a$ at $P_a$.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With the interlock mechanism inoperable:

1. Maintain at least one operable air lock door closed and either return the interlock to service within 24 hours or lock at least one operable air lock door closed.

2. Operation may then continue until the interlock is returned to service provided that one of the air lock doors is verified locked closed prior to each closing of the shield door and at least once per shift while the shield door is open.

3. Personnel passage through the air lock is permitted provided an individual is dedicated to assure that one operable air lock door remains locked at all times so that both air lock doors cannot be opened simultaneously.

4. The provisions of Specification 3.0.4 are not applicable.

b. With one primary 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 immediately prior to each closing of the shield door and at least once per shift while the shield door is open.

*See Special Test Exception 3.10.1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

c. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or an inoperable interlock mechanism, 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 SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying interlock operation (i.e., that only one door in each air lock can be opened at a time).

   1. Prior to using the air lock in Operating Conditions 1, 2 and 3 but not required more than once per 6 months,

   2. Following maintenance that could affect the interlock mechanism.

b. 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 rate less than or equal to 0.025 $L_a$ when the gap between the door seals is pressurized to 10 psig.

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

   1. At least once per 6 months###, and

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance had been performed on the air lock that could affect the air lock sealing capability*.

###The provisions of Specification 4.0.2 are not applicable.

* Exception to Appendix J of 10 CFR 50.

slightly increasing the time required for the normal scram to suppress the flux.

## 3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

## 3/4.2.6 POWER/FLOW INSTABILITY

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

In 1984, GE issued SIL 380 addressing boiling instability and made several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress." The SIL recommends that REGION A be bounded by the 100% rod line and REGION C be bounded by the 80% rod line.

The NRC Generic Letter 86-02 discussed both the GE and SIEMENS (then EXXON) stability methodology and stated that due to uncertainties, General Design Criteria 10 and 12 could not be met using available analytical procedures on a BWR. The letter discussed SIL 380 and stated that General Design Criteria 10 and 12 could be met by imposing SIL 380 recommendations in operating regions of potential instabilities. The NRC concluded that regions of potential instability constituted decay ratios of 0.8 and greater by the GE methodology and 0.75 by the SIEMENS methodology which existed at that time.

SIEMENS Power Corporation has recently developed an improved stability computer code STAIF. A topical report (EMF-CC-074P) which describes the STAIF stability code and provides benchmarking against reactor data was submitted to the NRC in 1993. The NRC issued a SER approving the STAIF stability code for establishing stability boundaries on April 14, 1994. In the SER on STAIF the NRC stated the uncertainty in the STAIF code was 20%.

The STAIF stability code has been used to establish the stability region boundaries for WNP-2. The lower boundary of REGION A was defined to assure it bounds a decay ratio of 0.9. REGION C was conservatively defined to bound a decay ratio of 0.75.

Stability REGION A is shown in Figure 3.2.6-1. REGION A conforms to the recommendations of SIL 380 in that REGION A bounds a calculated decay ratio of 0.9. Operation in REGION A is prohibited.

## 3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux signal decay ratios should be monitored while operating in this region (Region C).

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.75 was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power at which the calculated decay ratio is less than 0.75.

Stability monitoring is performed utilizing the ANNA system. The system shall be used to monitor APRM and LPRM signal decay ratio and peak-to-peak noise values when operating in the region of concern. A minimum number of LPRM and APRM signals are required to be monitored in order to assure that both global (in-phase) and regional (out-of-phase) oscillations are detectable. Decay ratios are calculated from 30 seconds worth of data at a sample rate of 10 samples/second. This sample interval results in some inaccuracy in the decay ratio calculation, but provides rapid update in decay ratio data. A decay ratio of 0.75 is selected as a decay ratio limit for operator response such that sufficient margin to an instability occurrence is maintained. When operating in the region of applicability, decay ratio and peak-to-peak information shall be continuously calculated and displayed. A surveillance requirement to continuously monitor decay ratio and peak-to-peak noise values ensures rapid response such that changes in core conditions do not result in approaching a point of instability.

## 3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

The basis for stability monitoring during single loop operation is consistent with that given above for two loop operation. The smaller size of the region of allowable operation, Region C, is due to a limit on the allowed flow above the 80% rodline. When operating above the 80% rodline in single loop operation, the core flow is required to be greater than 39%. Continuous operation in Region B is not permitted. Should Region B be entered the actions required by Technical Specification 3/4.4.1.1 are to be complied with.

## 3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady-state operation will not exceed small fractions of the dose guidelines of 10 CFR Part 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, 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 primary coolant's specific activity greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. 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.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$. Reactor operation and resultant fast neutron irradiation, E greater than 1 MeV, will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence, nickel content, and copper content of the material in question, can be predicted using the fluence for 109.2% of original rated power and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curves, Figures 3.4.6.1A, 3.4.6.1B, and 3.4.6.1C include predicted adjustments for this shift in $RT_{NDT}$ for the end of life fluence and are effective for 10 EFPY and 8 EFPY, respectively.

The actual shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1A, 3.4.6.1B, and 3.4.6.1C shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1A, 3.4.6.1B, and 3.4.6.1C for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

## 3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

## 3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.

## 3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.