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## 2.0 SAFETY LIMITS

## 2.1 SAFETY LIMITS

A. Reactor Core Safety Limits

1. With the reactor steam dome pressure $< 735$ psig or core flow $< 10\%$ rated core flow:

   Thermal power shall be $\leq 25\%$ Rated Thermal Power

2. With the reactor steam dome pressure $\geq 785$ psig and core flow $\geq 10\%$ rated core flow:

   MCPR shall be $\geq 1.10$ for two recirculation loop operation or $\geq 1.12$ for single recirculation loop operation.

3. Reactor vessel water level shall be greater than the top of active irradiated fuel.

B. Reactor Coolant System Pressure Safety Limit

Reactor steam dome pressure shall be $\leq 1332$ psig.

## LIMITING SAFETY SYSTEM SETTINGS

**Limiting Safety System Settings are incorporated into Section 3 of the Technical Specifications.**

| 2.0 SAFETY LIMITS | LIMITING SAFETY SYSTEM SETTINGS |
|---|---|
| **2.2 SAFETY LIMIT VIOLATIONS**<br><br>With any Safety Limit violation, the following actions shall be completed within 2 hours:<br><br>A. Restore compliance with all Safety Limits; and<br><br>B. Insert all insertable control rods. | |

Bases 2.1:

A. The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is no less than the values specified in Technical Specification 2.1.A. This limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection systems safety settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold, beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with margin to the conditions which would produce onset of transition boiling. (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. The concept of MCPR, as used in the GETAB/GEXL critical power analyses, is discussed in Reference 1.

1. Core Thermal Power Limit (Reactor Pressure $\leq$ 785 psig or Core Flow $\leq$ 10% of Rated) At pressure below 785 psig, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and all core flows, this pressure differential is maintained in the bypass region of the core.

   Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and all flows will always be greater than 4.56 psi. Analyses show that with a bundle flow of 28 x $10^3$ lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Therefore, due to the 4.56 psi driving head, the bundle flow will be greater than 28 x $10^3$ lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 0 to 785 psig indicate that the fuel assembly critical power at 28 x $10^3$ lbs/hr is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 785 psig or core flow less than 10% is conservative.

Bases 2.1 (Continued):

2. Core Thermal Power Limit (Reactor Pressure ≥785 psig and Core Flow ≥10% of Rated.) Onset of transition boiling results in a decrease in heat transfer from the clad and, therefore, elevated clad temperature and the possibility of clad failure. However, the existence of critical power, or boiling transition, is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR) which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. The Safety Limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the Operating MCPR Limit (T.S.3.11.C) more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the Safety Limit is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state including uncertainty in the boiling transition correlation as described in Reference 1. The uncertainties employed in deriving the Safety Limit are provided at the beginning of each fuel cycle.

   Because the boiling transition correlation is based on a large quantity of full scale data, there is a very high confidence that operation of a fuel assembly at the MCPR Safety Limit would not produce boiling transition. Thus, although it is not required to establish the Safety Limit, additional margin exists between the Safety Limit and the actual occurrence of loss of cladding integrity.

   However, if boiling transition were to occur, clad perforation would not be expected. Cladding temperatures would increase to approximately 1100°F which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR) where fuel similar in design to Monticello operated above the boiling transition for a significant period of time (30 minutes) without clad perforation.

   If reactor pressure should ever exceed 1385 psig during normal power operation (the limit of applicability of the boiling transition correlation) it would be assumed that the fuel cladding integrity Safety Limit has been violated.

   In addition to the MCPR Safety Limit, operation is constrained to a maximum design linear heat generation rate for any fuel type in the core.

3. Reactor Water Level (Shutdown Condition) During periods when the reactor is shut down, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. Establishment of the safety limit above the top of the fuel provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

Bases 2.1 (Continued):

B. The pressure safety limit of 1332 psig as measured in the vessel steam space was derived from the design pressures of the reactor pressure vessel, steam space piping, water space piping, and recirculation pump casing. The respective design pressures are 1250 psig, 1110 psig, 1136 psig, and 1380 psig. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code Section III-A for the pressure vessel, ASME Boiler and Pressure Vessel Code Section III-C for the recirculation pump casing, and USAS Piping Code Section B31.1 for the reactor coolant system piping. The ASME Code permits pressure transients up to 10% over the vessel design pressure (110% x 1250 = 1375 psig) and the USAS Code permits pressure transients up to 20% over the piping design pressure (120% x 1110 = 1332 psig for piping communicating with the vessel steam space and 120% x 1136 = 1363 psig at the bottom of the vessel). The pressure limit is 1332 psig based on reactor coolant system steam piping.

References

1. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO 10958.

Bases 2.2:

Exceeding a Safety Limit may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," guidelines. Therefore, it is required to insert all insertable control rods and restore compliance with the Safety Limits within 2 hours. The 2 hour completion time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal. Other required actions are delineated in 10 CFR 50.36, 10 CFR 50.72, and 10 CFR 50.73



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TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

| Trip Function | Limiting Trip Settings | Modes in which function must be Operable or Operating** | | | Total No. of Instrument Channels per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1) | Required Condition* |
|---|---|---|---|---|---|---|---|
| | | Refuel (3) | Startup | Run | | | |
| 1. Mode Switch in Shutdown | | X | X | X | 1 | 1 | A |
| 2. Manual Scram | | X | X | X | 1 | 1 | A |
| 3. Neutron Flux IRM (See Note 2)<br>a. High-High<br>b. Inoperative | ≤120/125 of full scale AND <20% of Rated Thermal Power | X | X | | 4 | 3 | A |
| 4. Flow Referenced Neutron Flux APRM (See Note 5)<br>a. High-High<br>b. Inoperative | ≤[0.66W+65.6] %Rated Thermal Power for two loop operation OR ≤[0.66(W-5.4)+65.6] %Rated Thermal Power for single loop operation<br><br>Where: W=percent of recirculation drive flow to produce a core flow of $57.6x10^6$ lbm/hr | | | X | 3 | 2 | A or B |
| c. High Flow Clamp | ≤120 % | | | | | | |
| 5. High Reactor Pressure (See Note 9) | ≤1075 psig | X | X(f) | X(f) | 2 | 2 | A |

| TABLE 3.1.1 - CONTINUED | | | | | | | |
|---|---|---|---|---|---|---|---|
| Trip Function | Limiting Trip Settings | Modes in which function must be Operable or Operating** | | | Total No. of Instrument Channels per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1) | Required Condition* |
| | | Refuel (3) | Startup | Run | | | |
| 6. High Drywell Pressure (See Note 4) | ≤2 psig | X | X(e, f) | X(e, f) | 2 | 2 | A |
| 7. Reactor Low Water Level | ≥7 in. | X | X(f) | X(f) | 2 | 2 | A |
| 8. Scram Discharge Volume High Level | | | | | | | |
| a. East | ≤56 gal. (8) | X(a) | X(f) | X(f) | 2 | 2 | A |
| b. West | ≤56 gal. (8) | X(a) | X(f) | X(f) | 2 | 2 | A |
| 9. Turbine Condenser Low Vacuum | ≥22 in. Hg | X(b) | X(b,f) | X(f) | 2 | 2 | A or C |
| 10. Main Steamline Isolation Valve Closure | ≤10% Valve Closure | X(b) | X(b) | X | 8 | 8 | A or C |
| 11. Turbine Control Valve Fast Closure | (See Note 7) | | | X(d, f) | 2 | 2 | D |
| 12. Turbine Stop Valve Closure | ≤10% Valve Closure | | | X(d) | 4 | 4 | D |

NOTES:

1. There shall be two operable or tripped trip systems for each function. 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 that at least one other operable channel in the same trip system is monitoring that parameter.
2. For an IRM channel to be considered operable, its detector shall be fully inserted.
3. In the refueling mode with the reactor subcritical and reactor water temperature less than 212°F, only the following trip functions need to be operable: (a) Mode Switch in Shutdown, (b) Manual Scram, (c) High Flux IRM, (d) Scram Discharge Volume High Level.
4. Not required to be operable when primary containment integrity is not required.
5. To be considered operable, an APRM must have at least 2 LPRM inputs per level and at least a total of 14 LPRM inputs, except that channels 1, 2, 5, and 6 may lose all LPRM inputs from the companion APRM Cabinet plus one additional LPRM input and still be considered operable.

Bases 3.1 (Continued):

1. Mode Switch in Shutdown

   A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status. Reference Section 7.6.1 of the USAR.

2. Manual Scram

   The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

3. Neutron Flux IRM Scram

   For operation in the startup mode while the reactor is at low pressure, the IRM scram setting of 20% of rated power provides adequate thermal margin between the setpoint and the safety limit, 25% of rated. The margin is adequate to accommodate anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position and the associated APRM is not downscale. This switch occurs when reactor pressure is greater than 850 psig.

   The IRMs are calibrated by the heat balance method such that 120/125 of full scale on the highest IRM range is below 20% of rated neutron flux. The requirement that the IRM detectors be inserted in the core assures that the heat balance calibration is not invalidated by the withdrawal of the detector.

4. Neutron Flux IRM Scram

Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1775 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that, with a 120% scram trip setting, none of the abnormal operational transients analyzed violate the fuel Safety Limit and there is a substantial margin from fuel damage. Also, the flow biased neutron flux scram provides protection to the fuel safety limit in the unlikely event of a thermal-hydraulic instability.

Maximum Extended Load Line Limit Analyses (MELLLA) have been performed to allow operation at higher powers at flows below 87%. The flow referenced scram (and rod block line) have increased (higher slope and y-intercept) for two loop operation (See Core Operating Limits Report). The supporting analyses are discussed in GE NEDC-31849P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Increased Core Flow (ICF) analyses have been performed to allow operating at flows above 100% for powers equal to or less than 100% (See Core Operating Limit Report). The supporting analyses are discussed in General Electric NEDC-31778P report (Reference: Letter from NSP to NRC dated September 16, 1992).

Evaluations discussed in NEDC-32546P, July 1996, demonstrate the operability of MELLLA and ICF for rerate conditions. In addition, the evaluation demonstrated the acceptability of MELLLA for single loop operation.

5. High Reactor Pressure Scram

The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

Bases 3.1 (Continued):

6. High Drywell Pressure Scram

Instrumentation (pressure switches) in the drywell are provided to detect a loss of coolant accident and initiate the emergency core cooling equipment. This instrumentation is a backup to the water level instrumentation which is discussed in Specification 3.2.

7. Reactor Low Water Level Scram

The low reactor water level instrumentation is set to trip when reactor water level is $\geq 7''$ on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected safety analyses. All Technical Specification reactor water level setpoints are specified as inches measured in the reactor annulus and referenced to instrument "zero." Instrument "zero" is a point 477.5″ above the inner clad surface on the bottom of the reactor vessel.

8. Scram Discharge Volume Scram

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by the scram can be accommodated in the discharge piping. Part of this piping consists of two instrument volumes which accommodate in excess of 56 gallons of water each and is the low point in the piping. During normal operation the discharge volumes are empty; however, should they fill with water, the water discharge to the piping from the reactor could not be accommodated which would result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been provided in the instrument volumes which alarm and scram the reactor when the volume of water in either of the discharge volume receiver tanks reaches 56 gallons. At this point there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

9. Turbine Condenser Low Vacuum

Loss of condenser vacuum occurs when the condenser can no longer handle the heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. The condenser low vacuum scram is a back-up to the stop valve closure scram and causes a scram before the stop valves are closed and thus the resulting transient is less severe. Scram occurs at 22″ Hg vacuum, stop valve closure occurs at 20″ Hg vacuum, and bypass closure at 7″ Hg vacuum.

10. Main Steamline Isolation Valve Closure

The main steamline isolation valve closure scram is set to scram when the isolation valves are ≤10% closed from full open. This scram anticipates the pressure and flux transient, which would occur when the valves close. By scramming at this setting the resultant transient is insignificant.

11. Turbine control Valve Fast Closure

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

12. Turbine Stop Valve Closure

The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed. Specific analyses have generated specific limits which allow this scram to be bypassed below 45% rated thermal power. In order to ensure the availability of this scram above 45% rated thermal power, this scram is only bypassed below 30% thermal power as indicated by turbine first stage pressure. This takes into account the possibility of 14% power being passed directly to the condenser through the bypass valves.

Although the operator will set the set points within the trip settings specified on Table 3.1.1, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. For power rerate, GE setpoint methodology provided in NEDC 31336, "General Electric Setpoint Methodology," is used in establishing setpoints. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, such deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts:

| Trip Function | Deviation | Trip Function | Deviation |
|---|---|---|---|
| 3. High Flux IRM | +2/125 of scale | *7. Reactor Low Water Level | -6 inches |
| 5. High Reactor Pressure | +10 psi | 8. Scram Discharge Volume High Level | +1 gallon |
| 6. High Drywell Pressure | +1 psi | 9. Turbine Condenser Low Vacuum | -1/2 in. Hg |

* This indication is reactor coolant temperature sensitive. The calibration is thus made for rated conditions. The level error at low pressures and temperatures is bounded by the safety analysis which reflects the weight-of-coolant above the lower tap, and not the indicated level.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable, or the actions specified in 3.1.B are not initiated as specified.

If an unsafe failure is detected during surveillance testing, it is desirable to determine as soon as possible if other failures of a similar type have occurred and whether the particular function involved is still operable or capable of meeting the single failure criterion. To meet the requirements of Table 3.1.1, it is necessary that all instrument channels in one trip system be operable to permit testing in the other trip system. Thus, when failures are detected in the first trip system tested, they would have to be repaired before testing of the other system could begin. In the majority of cases, repairs or replacement can be accomplished quickly. If repair or replacement cannot be completed in a reasonable time, operation could continue with one tripped trip system until the surveillance testing deadline.

The ability to bypass one instrument channel when necessary to complete surveillance testing will preclude continued operation with scram functions which may be either unable to meet the single failure criterion or completely inoperable. It also eliminates the need for an unnecessary shutdown if the remaining channels are found to be operable. The conditions under which the bypass is permitted require an immediate determination that the particular function is operable. However, during the time a bypass is applied, the function will not meet the single failure criterion; therefore, it is prudent to limit the time the bypass is in effect by requiring that surveillance testing proceed on a continuous basis and that the bypass be removed as soon as testing is completed.


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Table 3.2.1
Instrumentation That Initi es Primary Containment Isolation Functions

| Function | Trip Settings | Total No. of Instrument Channels Per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2) | Required Conditions* |
|---|---|---|---|---|
| 1. Main Steam and Recirc <u>Sample Line (Group 1)</u> | | | | |
| a. Low Low Reactor Water Level | ≥-48″ | 2 | 2 | A |
| b. High Flow In Main Steam Line | ≤140% rated | 8 | 8 | A |
| c. High temp. in Main Steam Line Tunnel | ≤200°F | 8 | 2 of 4 in each of 2 sets | A |
| d. Low Pressure in Main Steam Line (3) | ≥825 psig | 2 | 2 | B |
| 2. RHR System, Head Cooling, Drywell, Sump, TIP (Group 2) | | | | |
| a. Low Reactor Water Level | ≥7″ | 2 | 2 | C |

| Table 3.2.1 (Continued) | | | | |
|---|---|---|---|---|
| Function | Trip Settings | Total No. of Instrument Channels Per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (1, 2) | Required Conditions* |
| b. High Drywell Pressure (5) | ≤2 psig | 2 | 2 | D |
| 3. Reactor Cleanup System (Group 3) | | | | |
| a. High Drywell Pressure | ≤2 psig | 2 | 2 | E |
| b. Low Low Reactor Water Level** | ≥-48" | 2 | 2 | E |
| c. High RWCU Room Temperature Allowable Value | ≤188°F | 2 | 2 | E |
| d. High RWCU System Flow Allowable Value | ≤500 gpm with ≤27 second time delay | 2 | 2 | E |
| 4. HPCI Steam Lines (Group 4) | | | | |
| a. HPCI High Steam Flow*** | ≤300,000 lb/hr with ≤7 second time delay | 2(4) | 2 | F |
| b. HPCI Steam Line Area High Temp. | ≤200°F | 16(4) | 16 | F |
| c. Low Pressure in HPCI Steam Supply Line | ≥85 psig | 4(6) | 4(6) | F |

| Table 3.2.2<br>Instrumentation That Initiates Emergency Core Cooling Systems | | | | | |
|---|---|---|---|---|---|
| Function | Trip Setting | Minimum No. of Operable or Operating Trip Systems (3) (6) | Total No. of Instrument Channels Per Trip System | Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6) | Required Conditions* |
| A. <u>Core Spray and LPCI</u> | | | | | |
| 1. Pump Start | | | | | |
| a. Low Low Reactor Water Level and | ≥-48″ | 2 | 4(4) | 4 | A. |
| b. i. Reactor Low Pressure Permissive or | ≥450 psig | 2 | 2(4) | 2 | A. |
| ii. Reactor Low Pressure Permissive Bypass Timer | 20±1 min | 2 | 1 | 1 | B. |
| c. High Drywell Pressure (1) | ≤2 psig | 2 | 4(4) | 4 | A. |
| 2. Low Reactor Pressure (Valve Permissive) | ≥450 psig | 2 | 2(4) | 2 | A. |
| 3. Loss of Auxiliary Power | ------ | 2 | 2(2) | 2 | A. |

| Table 3.2.2<br>Instrumentation That Initiates Emergency Core Cooling Systems | | | | | |
|---|---|---|---|---|---|
| Function | Trip Setting | Minimum No. of Operable or Operating Trip Systems (3) (6) | Total No. of Instrument Channels Per Trip System | Minimum No. of Operable or Operating Instrument Channels Per Trip System (3) (6) | Required Conditions* |
| B. <u>HPCI System</u> | | | | | |
| 1. High Drywell Pressure (1) | ≤2 psig | 1 | 4 | 4 | A. |
| 2. Low-Low Reactor Water Level | ≥-48″ | 1 | 4 | 4 | A. |
| C. <u>Automatic Depressurization</u> | | | | | |
| 1. Low-Low Reactor Water Level and | ≥-48″ | 2 | 2 | 2 | B. |
| 2. Auto Blowdown Timer and | ≤120 seconds | 2 | 1 | 1 | B. |
| 3. Low Pressure Core Cooling Pumps Discharge Pressure Interlock | ≥60 psig<br>≤150 psig | 2 | 12(4) | 12(4) | B. |

| Table 3.2.2 - (Continued)<br>Instrumentation That Initiates Emergency Core Cooling Systems | | | | | |
|---|---|---|---|---|---|
| Function | Trip Setting | Minimum No. of Operable or Operating Trip Systems (3) (6) | Total No. of Instrument Channels Per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System (3) (6) | Required Conditions* |
| D. <u>Diesel Generator</u> | | | | | |
| 1. Degraded or Loss of Voltage Essential Bus (5) | | | | | |
| 2. Low Low Reactor Water Level | ≥-48" | 2 | 4(4) | 4 | C. |
| 3. High Drywell Press | ≤2 psig | 2 | 4(4) | 4 | C. |

<u>NOTES:</u>

1. High drywell pressure may be bypassed when necessary only by closing the manual containment isolation valves during purging for containment inerting or de-inerting. Verification of the bypass condition shall be noted in the control room log. Also need not be operable when primary containment integrity is not required.

2. One instrument channel is a circuit breaker contact and the other is an undervoltage relay.

Table 3.2.4
Instrumentation That Initiates Reactor Building Ventilation Isolation
And Standby Gas Treatment System Initiation

| Function | Trip Settings | Total No. of Instrument Channels Per Trip System | Min. No. of Operable or Operating Instrument Channels Per Trip System | Required Conditions* |
|---|---|---|---|---|
| 1. Low Low Reactor Water Level | ≥-48" | 2 | 2 (Notes 1, 3, 5, 6) | A. or B. |
| 2. High Drywell Pressure | ≤2 psig | 2 | 2 (Notes 1, 3, 5, 6) | A. or B. |
| 3. Reactor Building Plenum Radiation Monitors | ≤100 mR/hr | 1 | 1 (Notes 1, 2, 4) | A. or B. |
| 4. Refueling Floor Radiation Monitors | ≤100 mR/hr | 1 | 1 (Notes 1, 2, 4) | A. or B. |

Notes:

(1) There shall be two operable or tripped trip systems for each function with two instrument channels per trip system and there shall be one operable or tripped trip system for each function with one instrument channel per trip system.

(2) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied action shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(3) Need not be operable when primary containment integrity is not required.

(4) One of the two monitors may be bypassed for maintenance and/or testing.

Table 3.2.5
Instrumentation That Initiates a Recirculation Pump Trip
and Alternate Rod Injection

| Function | Trip Setting | Minimum No. of Operable or Operating Trip Systems (1) | Total No. of Instrument Channels per Trip System | Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) | Required Conditions* |
|---|---|---|---|---|---|
| 1. High Reactor Dome Pressure | ≤1150 psig | 2 | 2 | 2 | A |
| 2. Low-Low Reactor Water Level | ≥-48" | 2 | 2 | 2 | A |

NOTE:

1. When one of the two trip systems is made or found to be inoperable, restore the inoperable trip system to operable status within 14 days or place the plant in the specified required condition within the next eight hours. When both trip systems are inoperable, place the plant in the specified required condition within eight hours unless at least one trip system is sooner made operable.

* Required conditions when minimum conditions for operation are not satisfied:

A. Reactor in Startup, Refuel, or Shutdown Mode.

| Table 3.2.8<br>Other Instrumentation | | | | | |
|---|---|---|---|---|---|
| Function | Trip Setting | Minimum No. of Operable or Operating Trip System (1) (2) | Total No. of Instrument Channels Per Trip System | Minimum No. of Operable or Operating Instrument Channels Per Trip System (1) (2) | Required Conditions* |
| A. RCIC Initiation<br>1. Low-Low Reactor Level | ≥ -48″ | 1 | 4 | 4 | B |
| B. HPCI/RCIC Turbine Shutdown<br>1. High Reactor Level | ≤ 48″ | 1 | 2 | 2 | A |
| C. HPCI/RCIC Turbine Suction Transfer<br>1. Condensate Storage Tank Low Level Allowable Values | ≥ 2′ 3″ above tank bottom (Two Tank Operation) | 1 | 2 | 2 | C |
| | ≥ 6′ 9″ above tank bottom (One Tank Operation) | 1 | 2 | 2 | C |

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated as follows:
   a. With one required instrument channel inoperable per trip function, place the inoperable channel or trip system in the tripped condition within 12 hours, or
   b. With more than one instrument channel per trip system inoperable, immediately satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or
   c. Place the plant under the specified required condition using normal operating procedures.
2. 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 that at least one other operable channel in the same trip system is monitoring that parameter.

* Required conditions when minimum conditions for operation are not satisfied:

A. Comply with Specification 3.5.A.
B. Comply with Specification 3.5.D.
C. Align HPCI and RCIC suction to the suppression pool.

Bases 3.2:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is > 7″ on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of Group 2 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is ≥ -48″. This trip initiates closure of the Group 1 and Group 3 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generators.

| 3.0 LIMITING CONDITIONS FOR OPERATION | 4.0 SURVEILLANCE REQUIREMENTS |
|---|---|
| | |

F. Recirculation System

3. The reactor may be started and operated, or operation may continue with only one recirculation loop in operation provided that:

   a. The following changes to setpoints and safety limit settings will be made within 24 hours after initiating operation with only one recirculation loop in operation.

      1. The Operating Limit MCPR (MCPR) will be changed per Specification 3.11.C.
      2. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) will be changed per Specification 3.11.A.
      3. The APRM Neutron Flux Scram and APRM Rod Block setpoints will be changed as noted in Tables 3.1.1 and 3.2.3.

   b. Technical Specifications 3.5.F.1 and 3.5.F.2 are met.

4. With no reactor coolant system recirculation loops in operation:

   a. Comply with Technical Specifications 3.5.F.1 and 3.5.F.2 by inserting control rods and then comply with specifications 3.6.A.2 and 3.5.F.3 for operation with only one recirculation loop in operation,

      OR

   b. The reactor shall be placed in hot shutdown within 12 hours.

## 3.0 LIMITING CONDITIONS FOR OPERATION

E. Safety/Relief Valves

1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F the safety valve function (self actuation) of seven safety/relief valves shall be operable (note: Low-Low Set and ADS requirements are located in Specification 3.2.H. and 3.5.A, respectively).

   Valves shall be set as follows:

   8 valves at ≤ 1120 psig

2. If Specification 3.6.E.1 is not met, initiate an orderly shutdown and have reactor coolant pressure and temperature reduced to 110 psig or less and 345°F or less within 24 hours.

## 4.0 SURVEILLANCE REQUIREMENTS

E. Safety/Relief Valves

1. a. Safety/relief valves shall be tested or replaced each refueling outage in accordance with the Inservice Testing Program.

   b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage.

   c. The integrity of the safety/relief valve bellows shall be continuously monitored.

   d. The operability of the bellows monitoring system shall be demonstrated each operating cycle.

2. Low-Low Set Logic surveillance shall be performed in accordance with Table 4.2.1.

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

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MCPR Limit is determined from the analysis of transients discussed in Bases Section 2.1. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

Flow runout events are analyzed with the purpose of establishing a flow dependent MCPR limit that would prevent the Safety Limit CPR from being reached during a flow runout. A flow runout event is a slow flow and power increase which is not terminated by a scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Initial conditions for the transient are set such that the limiting CPR is near the Safety Limit. MCPR values are determined from the resulting change in CPR when core flow is increased to a possible maximum. Several combinations of initial power, flow, and exposure are analyzed to cover the range of operability defined by the power/flow map. The calculated flow dependent MCPR limit ($MCPR_f$) for a given core flow is provided in the Core Operating Limits Report.

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, provided in the Core Operating Limits Report. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are provided in the Core Operating Limits Report. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error.

6.2 (Deleted)

6.3 (Deleted)

6.4 (Deleted)

7. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

Rod Block Monitor Operability Requirements (Specification 3.2.C.2a)
Rod Block Monitor Upscale Trip Settings (Table 3.2.3, Item 4.a)
Recirculation System Power to Flow Map Stability Regions (Specification 3.5.F)
Maximum Average Planar Linear Heat Generation Rate Limits (Specification 3.11.A)
Linear Heat Generation Rate Limits (Specification 3.11.B)
Minimum Critical Power Ratio Limits (Specification 3.11.C)
Power to Flow Map (Bases 3.1)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (the approved version at the time the reload analyses are performed)

NSPNAD-8608-A, "Reload Safety Evaluation Methods for Application to the Monticello Nuclear Generating Plant" (the approved version at the time the reload analyses are performed)

NSPNAD-8609-A, "Qualification of Reactor Physics Methods for Application to Monticello" (the approved version at the time the reload analyses are performed)

NEDO-31960, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," June 1991 (the approved version at the time the reload analyses are performed)

NEDO-31960, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," March 1992 (the approved version at the time the reload analyses are performed)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The Core Operating Limits Report, including any mid-cycle revisions or supplements, shall be supplied upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.