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1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

1.2 DELETED

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle at that height.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel, including the required sensor, alarm, display, and trip functions, and shall include the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall be less than or equal to the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the CORE OPERATING LIMITS REPORT:

a. Once within 12 hours after THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER and at least once per 24 hours thereafter.

b. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or
b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a) Place the recirculation flow control system in the Local Manual mode, and
b) Reduce THERMAL POWER to ≤ 70% of RATED THERMAL POWER, and
c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and
d) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
e) Reduce the LINEAR HEAT GENERATION RATE (LHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.2.2; otherwise, with the Trip Setpoints and Allowable Values associated with one trip system not reduced to those applicable for single recirculation loop operation, place the affected trip system in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specifications 2.2.1 and 3.2.2.

3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 3.2.2 and 3.3.6; otherwise, with the Trip Setpoint and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel

*See Special Test Exception 3.10.4.

CORE OPERATING LIMITS REPORT (Continued)

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC as applicable in the following documents:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR-II)"

2. CENPD-397-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurement Technology"

The CORE OPERATING LIMITS REPORT will contain the complete identification for each of the TS referenced topical reports used to prepare the CORE OPERATING LIMITS REPORT (i.e., report number title, revision, date, and any supplements).

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

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, within the time period specified for each report.

6.9.3 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC 20555, with a copy to the USNRC Administrator, Region 1, via the Licensee Event Report System within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

SPECIAL REPORTS

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections,

RECORD RETENTION (Continued)

repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of inservice inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the Quality Assurance Program.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

RECORD RETENTION (Continued)

k. Records of SORC meetings and activities of the Nuclear Review Board (and activities of its predecessor, the Offsite Safety Review (OSR) staff).

l. Records of the snubber service life monitoring pursuant to Technical Specification 4.7.5.

m. Records of analyses required by the radiological environmental monitoring program which would permit evaluation of the accuracy of the analyses at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATIONAL MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.15 TECHNICAL SPECIFICATION (TS) BASES CONTROL PROGRAM

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. PSEG may make changes to the Bases without prior NRC approval provided the changes do not require either of the following:

   1. A change in the TS incorporated in the License, or

   2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. Proposed changes to the Bases that require either condition of Specification 6.15.b above shall be reviewed and approved by the NRC prior to implementation.

d. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

e. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and in the procedures used to calculate critical power. Calculation of the Safety Limit MCPR is defined in Reference 1.

Reference:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (The approved revision at the time the reload analyses are performed. The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.)

3/4.1.1 SHUTDOWN MARGIN

SHUTDOWN MARGIN (SDM) requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions, transients, and Design Bases Events;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SDM can be demonstrated by using solely analytical methods or by performing a test. SDM can be measured only by performing a test. A test involves collecting data with the reactor at a specified condition or series of conditions. The primary purpose of a SDM Demonstration is to ensure that SDM is equal to or greater than the SDM Limit for a specific core exposure. The primary purpose of a SDM Measurement is to provide SDM in % delta k/k that can be used for: 1) ensuring that SDM is equal to or greater than the SDM Limit for a range of core exposures, 2) determining the need for additional SDM Measurements during the cycle, 3) providing a benchmark for the core design (design vs. actual SDM), and 4) providing a benchmark for potential future analysis of SDM for such events as control rods incapable of full insertion. This higher level of application requires that a SDM Measurement is determined from testing and not through solely analytical methods. Since a SDM Measurement satisfies the primary purpose of a SDM Demonstration, it can be considered a special type of SDM Demonstration.

All SDM Demonstrations involve some usage of analytical methods. The performance of tests lessens the usage of analytical methods, reduces the uncertainty in the results, and thus requires a smaller SDM Limit needed to show adequate SDM. At one end of the spectrum is a series of local criticals where both SDM and the highest worth control rod are determined by test. Although this technique has the minimum uncertainty and thus has the smallest SDM Limit, it still uses analytical methods to determine the worth of all the other control rods. At the other end of the spectrum is usage of solely analytical methods prior to core verification. This technique has the maximum uncertainty and thus has the largest SDM Limit.

The SDM Limit must be increased if the highest worth control rod is determined solely analytically versus a test using the reactor (requires a series of local criticals). This higher limit accounts for uncertainties in the calculation of the highest worth control rod.

SDM is demonstrated to satisfy a variety of OPCON 5 surveillances at the beginning of each cycle and, if necessary, at any future entry to OPCON 5 during the cycle if the assumptions of the previous SDM Demonstration are no longer valid. In most situations, the SDM Demonstration will be based solely on analytical methods and a test will not be performed. If SDM is demonstrated by using solely analytical methods, then SDM must be adjusted to account for

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than 10% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to 10% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in Reference 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods. Operability of a RBM channel is assured for a given control rod when ≥ 50% of the LPRM inputs for each detector level are available for that rod. When < 50% of the LPRM inputs on either detector level are available, a case-by-case evaluation of channel operability is required.

rate, solution concentration or boron equivalent to meet the ATWS Rule must not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron control equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution (natural boron enrichment)."

The described minimum system parameters (82.4 gpm, 13.6 percent concentration and natural boron equivalent) will ensure an equivalent injection capability that exceeds the ATWS Rule requirement. The stated minimum allowable pumping rate of 82.4 gallons per minute is met through the simultaneous operation of both pumps.

The standby liquid control system will also provide the capability to raise and maintain the long-term post-accident coolant inventory pH levels to 7 or above. This will prevent significant fractions of the dissolved iodine from being converted to elemental iodine and then re-evolving to the containment atmosphere.

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications in this section help assure that the fuel can be operated safely and reliably during normal operation. In addition, the limits specified in these specifications help ensure that the fuel does not exceed specified safety and regulatory limits during anticipated operational occurrences and design basis accidents. Specifically, these limits:

1. Ensure that the limits specified in 10CFR50.46 are not exceeded following the postulated design basis loss of coolant accident.

2. Ensure reactor operations remains within licensed, analyzed power/flow limits.

3. Ensure that the MCPR Safety Limit is not violated following any anticipated operational occurrence.

4. Ensure fuel centerline temperatures remain below the melting temperature and peak cladding strain remains below 1% during steady state operation.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is a measure of the average Linear Heat Generation Rate (LHGR) of all the fuel rods in a fuel assembly at any axial location. The Technical Specification APLHGR is the LHGR of the highest-powered fuel rod assumed in the LOCA analyses divided by an assumed conservatively small local peaking factor. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded. The limiting value for the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. The calculation procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The post LOCA peak cladding temperature (PCT) is primarily a function of the APLHGR and is dependent only secondarily on the rod to rod power distribution within an assembly. The analytical models used in evaluating the postulated loss-of-coolant accidents are described in Reference 1. These models are consistent with the requirements of Appendix K to 10CRF50.

The exposure dependent APLHGR limits are reduced by an APLHGR multiplier (MAPFAC) at various operating conditions to ensure that all fuel design criteria are met for normal operation and LOCA.

For plant operation with single recirculation loop, a lower value for the APLHGR limit is specified in the CORE OPERATING LIMITS REPORT. This lower value accounts for an earlier transition from nucleate boiling which occurs following a loss-of-coolant accident in the single loop operation compared to two loop operation.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-upscale scram setting

APRM SETPOINTS (Continued)
and the flow biased neutron flux-upscale control rod block trip setpoints must be adjusted to ensure that the MCPR does not become less than the fuel cladding Safety Limit or that > 1% plastic strain does not occur in the degraded situation. The scram setpoints and rod block setpoints are adjusted in accordance with the formula in Specification 3.2.2 whenever it is known that the existing power distribution would cause the design LHGR to be exceeded at RATED THERMAL POWER.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.2.3 is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow (K_f) to ensure adherence to fuel design limits during the worst transient with moderate frequency that is postulated in Chapter 15.

Flow dependent MCPR limits (K_f) are determined by steady state methods using a core thermal hydraulic code (Reference 1). K_f curves are provided based on the maximum credible flow runout transient (i.e., runout of both loops).

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump-speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting

MINIMUM CRITICAL POWER RATIO (Continued)

control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. This specification assures that the Linear Heat Generation Rate (LHGR) in any fuel rod is less than the design linear heat generation even if fuel pellet densification is postulated. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs), and to ensure that the peak clad temperature (PCT) during postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during normal operation or the anticipated operational occurrences identified in Reference 1.

The analytical methods and assumptions used in evaluating the fuel system design limits are presented in Reference 1. The analytical methods and assumptions used in evaluating AOOs and normal operation that determine the LHGR limits are presented in Reference 1.

LHGR limits are developed as a function of exposure to ensure adherence to fuel design limits during the limiting AOOs. The exposure dependent LHGR limits are reduced by an LHGR multiplier (LHGRFAC) at various operating conditions to ensure that all fuel design criteria are met for normal operation and AOOs. A complete discussion of the analysis code is provided in Reference 2.

For single recirculation loop operation, the LHGRFAC multiplier is limited to a maximum value as given in the CORE OPERATING LIMITS REPORT. This maximum limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

References:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version).

2. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," August 1986, and NEDE-24154-P-A, Supplement 1, Volume 4, Revision 1, February 2000.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single loop operation is permitted if the MCPR fuel cladding Safety Limit is increased as noted by Specification 2.1.2, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2 respectively. APLHGR limits are decreased by the factor given in the CORE OPERATING LIMITS REPORT (COLR), LHGR limits are decreased by the factor given in the COLR,and MCPR operating limits are adjusted as specified in the COLR.

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 38% THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during the extended operation of the single recirculation loop mode.

An inoperable jet pump is not in itself a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core, thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation loop flow mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference > 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, 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 pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

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 0.6 was chosen as the bases 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 greater than that specified in Figure 3.4.1.1-1.

Plant specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant to plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1-12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of BWR fuel and are found to be of negligible consequence. In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5-10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three (3) times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow end of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline date taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operates to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 13 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case transient.

Demonstration of the safety relief valve lift settings occurs only during shutdown. The safety relief valve pilot stage assemblies are set pressure tested in accordance with the recommendations of General Electric SIL No. 196, Supplement 14 (April 23, 1984), "Target Rock 2-Stage SRV Set-

Point Drift." Set pressure tests of the safety relief valve main (mechanical) stage are conducted at least once every 5 years.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973 and Generic Letter 88-01, "NRC Position on IGSCC in BWR Austinitic Stainless Steel Piping."

Proceduralized, manual quantitative monitoring and calculation of leakage rates, found by the NRC staff, in GL 88-01, Supp. 1, to be an acceptable alternative during repair periods of up to 30 days, should be demonstrated to have accuracy comparable to the installed drywell floor and equipment drain sump monitoring system.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates 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 system 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, the reactor will be shutdown to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The limit placed upon the rate of increase in UNIDENTIFIED LEAKAGE meets the guidance of Generic Letter 88-01, "NRC Position on IGSCC in BWR Austinitic Stainless Steel Piping."

3/4.4.4 This section has been deleted.