BASES

SECTION PAGE

INSTRUMENTATION (Continued)

3/4.3 INSTRUMENTATION

3/4.3.11 OSCILLATION POWER RANGE MONITOR

LIMITING CONDITION FOR OPERATION

3.3.11 Four channels of the OPRM instrumentation shall be OPERABLE*. Each OPRM channel period based algorithm amplitude trip setpoint (Sp) shall be less than or equal to the Allowable Value as 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.

ACTIONS

a. With one or more required channels inoperable:
   1. Place the inoperable channels in trip within 30 days, or
   2. Place associated RPS trip system in trip within 30 days, or
   3. Initiate an alternate method to detect and suppress thermal hydraulic instability oscillations within 30 days.

b. With OPRM trip capability not maintained:
   1. Initiate alternate method to detect and suppress thermal hydraulic instability oscillations within 12 hours, and
   2. Restore OPRM trip capability within 120 days.

c. Otherwise, reduce THERMAL POWER to less than 25% RTP within 4 hours.

SURVEILLANCE REQUIREMENTS

4.3.11.1 Perform CHANNEL FUNCTIONAL TEST at least once per 184 days.

4.3.11.2 Calibrate the local power range monitor once per 1000 Effective Full Power Hours (EFPH) in accordance with Note f, Table 4.3.1.1-1 of TS 3/4.3.1.

4.3.11.3 Perform CHANNEL CALIBRATION once per 18 months. Neutron detectors are excluded.

4.3.11.4 Perform LOGIC SYSTEM FUNCTIONAL TEST once per 18 months.

4.3.11.5 Verify OPRM is enabled when THERMAL POWER is ≥ 30% RTP and recirculation drive flow ≤ the value corresponding to 60% of rated core flow once per 18 months.

4.3.11.6 Verify the RPS RESPONSE TIME is within limits. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system. Neutron detectors are excluded.

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* When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated ACTIONS may be delayed for up to 6 hours, provided the OPRM maintains trip capability.

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.

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.

ACTION (Continued)

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 3.2.2 and 3.3.6.

4. Within 4 hours, reduce the Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoints and Allowable Values associated with one trip function not reduced to those applicable for single recirculation loop operation, place at least one affected channel in the tripped condition and within the following 6 hours, reduce the Trip setpoints and Allowable Values of the remaining channels to those applicable for single recirculation loop operation per Specification 3.3.6.

5. The provisions of Specification 3.0.4 are not applicable.

6. Otherwise be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loops in operation, initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation at least once per 12 hours verify that:

a. Reactor THERMAL POWER is ≤ 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is less than or equal to 90% of rated pump speed.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements or Specifications 4.4.1.1.2b and 4.4.1.1.2c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.3 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 109% and 107%, respectively, of rated core flow, at least once per 18 months.

FIGURE 3.4.1.1-1

DELETED

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the USNRC Administrator, Region 1, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the PSEG Nuclear LLC generated CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following Technical Specifications:

3/4.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE
3/4.2.3	MINIMUM CRITICAL POWER RATIO
3/4.2.4	LINEAR HEAT GENERATION RATE
3/4.3.11	OSCILLATION POWER RANGE MONITOR (OPRM)

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"

3. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, August 1996

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, 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.

RECORD RETENTION (Continued)

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.

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.

3/4.3.11 Oscillation Power Range Monitor (OPRM)

BACKGROUND

General Design Criterion 10 (GDC 10) requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Additionally, GDC 12 requires the reactor core and associated coolant, control, and protection systems to be designed to assure that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The OPRM System provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel minimum critical power ratio (MCPR) safety limit.

References 1, 2, and 3 describe three separate algorithms for detecting stability related oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. The OPRM System hardware implements these algorithms in microprocessor based modules. These modules, installed in local power range monitor (LPRM) flux amplifier slots in the Neutron Monitoring System (NMS) cabinets, execute the algorithms based on LPRM inputs and generate alarms and trips based on these calculations. These trips result in tripping the Reactor Protection System (RPS) when the appropriate RPS trip logic is satisfied, as described in the Bases for Specification 3.3.1, "RPS Instrumentation." Only the period based detection algorithm is used in the safety analysis. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations.

The period based detection algorithm detects a stability related oscillation based on the occurrence of a fixed number of consecutive LPRM signal period confirmations followed by the LPRM signal amplitude exceeding a specified setpoint. Upon detection of a stability related oscillation, a trip is generated for that OPRM channel.

The OPRM System consists of 4 OPRM trip channels, each channel consisting of two OPRM modules. Each OPRM module receives input from LPRMs. Each OPRM module also receives input from the RPS average power range monitor (APRM) power and flow signals to automatically enable the trip function of the OPRM module.

Each OPRM module is continuously tested by a self-test function. On detection of any OPRM module failure, either a Trouble alarm or INOP alarm is activated. The OPRM module provides an INOP alarm when the self-test feature indicates that the OPRM module may not be capable of meeting its functional requirements.

The OPRM System generates alarms based on system status and on the detection algorithms. However, this LCO specifies OPERABILITY requirements only for the period-based algorithm trip function.

APPLICABLE SAFETY ANALYSES

It has been shown that BWR cores may exhibit thermal-hydraulic reactor instabilities in high power and low flow portions of the core power to flow operating domain. GDC 10 requires the reactor core and associated coolant,

3/4.3.11 Oscillation Power Range Monitor (OPRM)

APPLICABLE SAFETY ANALYSES (continued)

control, and protection systems to be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC 12 requires assurance that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The OPRM System provides compliance with GDC 10 and GDC 12 by detecting the onset of oscillations and suppressing them by initiating a reactor scram. This assures that the MCPR safety limit will not be violated for anticipated oscillations.

The OPRM Instrumentation satisfies Criteria 3 of the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132).

LCO

Four channels of the OPRM System are required to be OPERABLE to ensure that stability related oscillations are detected and suppressed prior to exceeding the MCPR safety limit. Only one of the two OPRM modules' period based detection algorithms is required for OPRM channel OPERABILITY. The highly redundant and low minimum number of required LPRMs in the OPRM cell design ensures that large numbers of cells will remain operable, even with large numbers of LPRMs bypassed.

APPLICABILITY

The OPRM instrumentation is required to be OPERABLE in order to detect and suppress neutron flux oscillations in the event of thermal-hydraulic instability. As described in References 1, 2, and 3, the region of anticipated oscillation is defined by THERMAL POWER ≥ 30% RTP and recirculation drive flow ≤ the value corresponding to 60% of rated core flow. Therefore, the OPRM trip is required to be enabled in this region. However, to protect against anticipated transients, the OPRM is required to be OPERABLE with THERMAL POWER ≥ 25% RTP. This provides sufficient margin to potential instabilities as a result of a loss of feedwater heater transient. It is not necessary for the OPRM to be OPERABLE with THERMAL POWER < 25% RTP because instabilities are not anticipated to result from any expected transients below this power.

ACTIONS

a.1, a.2 and a.3

Because of the reliability and on-line self-testing of the OPRM instrumentation and the redundancy of the RPS design, an allowable out of service time of 30 days has been shown to be acceptable (Ref. 7) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the OPRM instrumentation still maintains OPRM trip capability (refer to Required Action b.1). The remaining OPERABLE OPRM channels continue to provide trip capability and provide operator information relative to stability activity. The remaining OPRM modules have high reliability. With this high reliability, there is a low probability of a subsequent channel failure within the allowable out of

3/4.3.11 Oscillation Power Range Monitor (OPRM)

ACTIONS (continued)

service time. In addition, the OPRM modules continue to perform on-line self-testing and alert the operator if any further system degradation occurs.

If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the OPRM channel or associated RPS trip system must be placed in the tripped condition per required actions a.1 and a.2. Placing the inoperable OPRM channel in trip (or the associated RPS trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the OPRM channel (or RPS trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), the alternate method of detecting and suppressing thermal-hydraulic instability oscillations is required (Required Action a.3). This alternate method is described in Reference 5. It consists of increased operator awareness and monitoring for neutron flux oscillations when operating in the region where oscillations are possible. If indications of oscillation, as described in Reference 5, are observed by the operator, the operator will take the actions described by procedures, which include initiating a manual scram of the reactor.

b.1 and b.2

Required action b.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped OPRM channels within the same RPS trip system result in not maintaining OPRM trip capability. OPRM trip capability is considered to be maintained when sufficient OPRM channels are OPERABLE or in trip (or the associated RPS trip system is in trip), such that a valid OPRM signal will generate a trip signal in both RPS trip systems. This would require both RPS trip systems to have one OPRM channel OPERABLE or in trip (or the associated RPS trip system in trip).

Because of the low probability of the occurrence of an instability, 12 hours is an acceptable time to initiate the alternate method of detecting and suppressing thermal-hydraulic instability oscillations described in Reference 5. The alternate method of detecting and suppressing thermal-hydraulic instability oscillations would adequately address detection and mitigation in the event of instability oscillations. Based on industry operating experience with actual instability oscillation, the operator would be able to recognize instabilities during this time and take action to suppress them through a manual scram. In addition, the OPRM System may still be available to provide alarms to the operator if the onset of oscillations were to occur. Since plant operation is minimized in areas where oscillations may occur, operation for 120 days without OPRM trip capability is considered acceptable with implementation of the alternate method of detecting and suppressing thermal-hydraulic instability oscillations.

c

With any required ACTION not met within the specified time interval, THERMAL POWER must be reduced to < 25% RTP within 4 hours. Reducing THERMAL POWER to < 25% RTP places the plant in a region where instabilities cannot occur. The 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER < 25% RTP from full power conditions in an orderly manner and without challenging plant systems.

3/4.3.11 Oscillation Power Range Monitor (OPRM)

SURVEILLANCE REQUIREMENTS

SR 4.3.11.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 184 days provides an acceptable level of system average availability over the Frequency and is based on the reliability of the channel (Ref. 7).

SR 4.3.11.2

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the OPRM System. The 1000 EFPH Frequency is based on operating experience with LPRM sensitivity changes. This surveillance is satisfied in accordance with Note f, Table 4.3.1.1-1 of TS 3/4.3.1.

SR 4.3.11.3

The CHANNEL CALIBRATION is a complete check of the instrument loop. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations, consistent with the plant specific setpoint methodology.

Calibration of the channel provides a check of the internal reference voltage and the internal processor clock frequency. It also compares the desired trip setpoints with those in processor memory. Since the OPRM is a digital system, the internal reference voltage and processor clock frequency are, in turn, used to automatically calibrate the internal analog to digital converters. The Allowable Values are specified in the CORE OPERATING LIMITS REPORT.

As noted, neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 1000 EFPH LPRM calibration using the TIPs (SR 4.3.11.2).

The Frequency of 18 months is based upon the design objective that the OPRM operate over a complete fuel cycle, as a minimum, without requiring calibration.

SR 4.3.11.4

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods and scram discharge volume (SDV) vent and drain valves in Specification 3.1.3.1, "Control Rod OPERABILITY" overlaps this Surveillance to provide complete testing of the assumed safety function. The OPRM self-test function may be utilized to perform this testing for those components that it is designed to monitor.

The 18-month Frequency is based on engineering judgment and reliability of the components. Operating experience has shown that these components usually pass the surveillance when performed at the 18 month Frequency.

3/4.3.11 Oscillation Power Range Monitor (OPRM)

SURVEILLANCE REQUIREMENTS (continued)

SR 4.3.11.5

This SR ensures that trips initiated from the OPRM system are not inadvertently bypassed when the capability of the OPRM system to initiate an RPS trip is required. The trip capability of the OPRM system is only required during operation under conditions susceptible to anticipated T-H instability oscillations. The region of anticipated oscillation is defined by THERMAL POWER ≥ 30% RTP and recirculation drive flow ≤ the value corresponding to 60% of rated core flow.

The trip capability of individual OPRM modules is automatically enabled based on the APRM power and flow signals associated with each OPRM channel during normal operation. These channel specific values of APRM power and recirculation drive flow are subject to surveillance requirements associated with other RPS functions such as APRM flux and flow biased simulated thermal power with respect to the accuracy of the signal to the process variable. The OPRM is a digital system with calibration and manually initiated tests to verify digital input including input to the auto-enable calculations. Periodic calibration confirms that the auto-enable function occurs at appropriate values of APRM power and recirculation flow signal. Therefore, verification that OPRM modules are enabled at any time that THERMAL POWER ≥ 30% RTP and recirculation drive flow ≤ the value corresponding to 60% of rated core flow adequately ensures that trips initiated from the OPRM system are not inadvertently bypassed.

The trip capability of individual OPRM modules can also be enabled by placing the module in the non-bypass (Manual Enable) mode. If placed in the non-bypass or Manual Enable mode the trip capability of the module is enabled and this SR is met. The frequency of 18 months is based on engineering judgment and reliability of the components.

SR 4.3.11.6

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis (Ref. 8). The OPRM self-test function may be utilized to perform this testing for those components it is designed to monitor. The RPS RESPONSE TIME acceptance criteria are included in Reference 8.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. RPS RESPONSE TIME tests are conducted such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system. This Frequency is based upon operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

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3/4.3.11 Oscillation Power Range Monitor (OPRM)

REFERENCES

1. NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995.

2. NEDO 31960-A, Supplement 1, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995.

3. NRC Letter, A. Thadani to L. A. England, "Acceptance for Referencing of Topical Reports NEDO-31960 and NEDO-31960, Supplement 1, 'BWR Owners Group Long-Term Stability Solutions Licensing Methodology'," July 12, 1993.

4. Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," July 11, 1994.

5. BWROG Letter BWROG-94078, "Guidelines for Stability Interim Corrective Action," June 6, 1994.

6. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Application," August 1996.

7. CENPD-400-P, Rev 01, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM)," May 1995.

8. GE-NE-A13-00381-04, "Reactor Long-Term Stability Solution Option III: Licensing Basis Hot Bundle Oscillation Magnitude for Hope Creek, " March 1999.

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.

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.