United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/97-0255
6 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
DEPARTURE FROM NUCLEATE BOILING CORRELATION

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS
AND BASES PAGES

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland WA 99352.

19. EMF-92-081(A), latest Revision and Supplements, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation - Nuclear Division, Richland, WA 99352.

20. EMF-92-153(P)(A), latest Revision and Supplements, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Richland WA 99352.

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens ~~ANFP~~ HTP correlation has a DNBR safety limit of ~~1.154~~ 1.141 (Ref. 3).

The Reactor Trip System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, flow, core power distribution, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Overtemperature ΔT trip;

b. Overpower ΔT trip;

c. Power Range Neutron Flux trip; and

d. Main steam safety valves.

Maintaining the DNBR above the limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The safety limit curves provided in Figure 2.1.1-1 remain valid using the Siemens HTP correlation.

BASES (continued)

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the unit in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

2. ~~XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."~~

3. ~~ANF-1224(P) Rev. 0, "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."~~

4. UFSAR, Sections 3.1, 4.4, 7.2, and 15.0.

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3. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Richland WA 99352.

XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, September 1983.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to ~~1.154~~ 1.141 using the ~~Advanced Nuclear Fuels Corporation's~~ Siemens Power Corporation's (SPC's) DNB

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BASES

BACKGROUND (continued)

correlation ~~(i.e., ANFP)~~ (i.e., HTP) and 1.17 using SPC's XNB correlation. All DNB limited transient events are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. Fuel design limits required by HBRSEP Design Criteria (Ref. 3) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of ~~1.254~~ 1.141 using the ~~ANFP~~ HTP correlation or 1.17 using the XNB correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs)

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1.141

BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion of ≥ 1.17 for the Standard Mixing Vane fuel, and ≥ ~~1.154~~ for the High Thermal Performance fuel (Ref. 2). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2205 psig and the RCS average temperature limit of 579.4°F correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 2.6% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all

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United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/97-0255
7 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE
DEPARTURE FROM NUCLEATE BOILING CORRELATION

RETYPED TECHNICAL SPECIFICATIONS AND BASES

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland WA 99352.

19. EMF-92-081(A), latest Revision and Supplements, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation - Nuclear Division, Richland, WA 99352.

20. EMF-92-153(P)(A), latest Revision and Supplements, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Power Corporation, Richland WA 99352.

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. The minimum DNB ratio, or DNBR, during normal operational and anticipated transients, is restricted to the safety limit. A DNBR at the safety limit corresponds to a 95% probability, at a 95% confidence level, that DNB will not occur, and is chosen as an appropriate margin to DNB for all operating conditions. The DNBR safety limit is a conservative design value which is used as a basis for setting core safety limits. Based on rod bundle tests, no fuel damage is expected at this DNBR or greater. For the standard mixing vane fuel, the Siemens Power Corporation XNB correlation has a DNBR safety limit of 1.17 (Ref. 2) and for the high thermal performance fuel the Siemens HTP correlation has a DNBR safety limit of 1.141 (Ref. 3). The safety limit curves provided in Figure 2.1.1-1 remain valid using the Siemens HTP correlation.

The Reactor Trip System setpoints specified in Limiting Condition for Operations (LCO) 3.3.1, in combination with all the LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System (RCS) temperature, pressurizer pressure, flow, core power distribution, and THERMAL POWER level that would result in a departure from nucleate boiling ratio (DNBR) of less than the DNBR limit and preclude the existence of flow instabilities.

Automatic enforcement of these reactor core SLs is provided by the following functions:

a. Overtemperature ΔT trip;

b. Overpower ΔT trip;

c. Power Range Neutron Flux trip; and

d. Main steam safety valves.

Maintaining the DNBR above the limit ensures that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid and also ensures that the ΔT

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BASES

APPLICABLE SAFETY ANALYSES (continued)

measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the Updated Final Safety Analysis Report (UFSAR), Ref. 4) provide more restrictive limits to ensure that the SLs are not exceeded.

SAFETY LIMITS

The curves provided in Figure 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and reactor vessel inlet temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the core exit quality is within the limits defined by the DNBR correlation. Figure 2.1.1-1 shows the allowable power level decreasing with increasing reactor vessel inlet temperature at selected pressurizer pressures for constant flow (i.e., three loop operation, minimum flow 97.3 x 10^6 lbm/hr). The area where clad integrity is assured is below these lines. The temperature limits at low power are considerably more conservative than would be required if they were based on the minimum allowable DNB ratio, but are set to preclude bulk boiling at the vessel exit. The safety limit curves given in Figure 2.1.1-1 are for constant flow conditions. These curves would not be applicable in cases where total reactor coolant flow is less than 97.3 x 10^6 lbm/hr. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the UFSAR.

The SL is higher than the limit calculated when the Axial Flux Difference (AFD) is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature and overpower ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Ref. 4).

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BASES (continued)

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The main steam safety valves and automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

SAFETY LIMIT VIOLATIONS

If SL 2.1.1 is violated, the requirement to restore compliance and go to MODE 3 places the unit in a safe condition and in a MODE in which this SL is not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. 10 CFR 50, Proposed Appendix A, 32FR10213, July 11, 1967.

2. XN-NF-621(P)(A) Revision 1, "Exxon Nuclear DNB Correlation PWR Fuel Designs," Exxon Nuclear Company, September 1983.

3. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."

4. UFSAR, Sections 3.1, 4.4, 7.2, and 15.0.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

BASES

BACKGROUND

The purpose of this LCO is to establish limits on the power density at any point in the core so that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. The design limits on local (pellet) and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$ is defined as the ratio of the integral of the linear power along the fuel rod with the highest integrated power to the average integrated fuel rod power. Therefore, $F^N_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod.

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least every 31 EFPD. However, during power operation, the global power distribution is monitored by LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD) (PDC-3 Axial Offset Control Methodology)," and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)," which address directly and continuously measured process variables.

The COLR provides peaking factor limits that ensure that the design basis value of the departure from nucleate boiling (DNB) is met for normal operation, operational transients, and any transient condition arising from events of moderate frequency. The DNB design basis precludes DNB and is met by limiting the minimum local DNB heat flux ratio to 1.141 using the Siemens Power Corporation's (SPC's) DNB correlation (i.e., HTP) and 1.17 using SPC's XNB

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BASES

BACKGROUND (continued)

correlation. All DNB limited transient events are assumed are assumed to begin with an $F^N_{\Delta H}$ value that satisfies the LCO requirements. Operation outside the LCO limits may produce unacceptable consequences if a DNB limiting event occurs. The DNB design basis ensures that there is no overheating of the fuel that results in possible cladding perforation with the release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSES

Limits on $F^N_{\Delta H}$ preclude core power distributions that exceed the following fuel design limits:

a. There must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hottest fuel rod in the core does not experience a DNB condition (Ref. 1);

b. During a large break loss of coolant accident (LOCA), peak cladding temperature (PCT) must not exceed 2200°F;

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 280 cal/gm (Ref. 2); and

d. Fuel design limits required by HBRSEP Design Criteria (Ref. 3) for the condition when control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn.

For transients that may be DNB limited, the Reactor Coolant System flow and $F^N_{\Delta H}$ are the core parameters of most importance. The limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met for normal operation, operational transients, and any transients arising from events of moderate frequency. The DNB design basis is met by limiting the minimum DNBR to the 95/95 DNB criterion of 1.141 using the HTP correlation or 1.17 using the XNB correlation. This value provides a high degree of assurance that the hottest fuel rod in the core does not experience a DNB.

The allowable $F^N_{\Delta H}$ limit increases with decreasing power level. This functionality in $F^N_{\Delta H}$ is included in the analyses that provide the Reactor Core Safety Limits (SLs)

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BASES

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion of ≥ 1.17 for the Standard Mixing Vane fuel, and ≥ 1.141 for the High Thermal Performance fuel (Ref. 2). This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

The pressurizer pressure limit of 2205 psig and the RCS average temperature limit of 579.4°F correspond to analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables – pressurizer pressure, RCS average temperature, and RCS total flow rate – to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error of 2.6% based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all

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