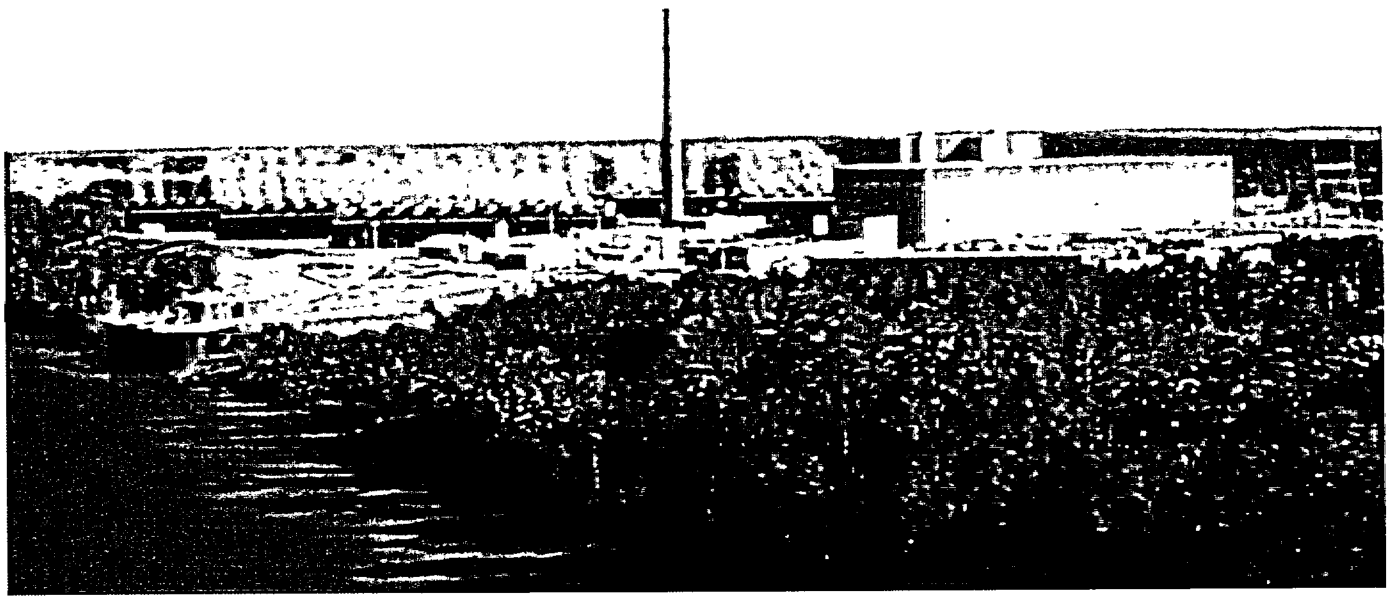
Edwin I. Hatch Nuclear Plant

Measurement Uncertainty Recapture Power Uprate



Licensing Submittal

December 2002

SOUTHERN COMPANY

Energy to Serve Your World®

General Electric Company

AFFIDAVIT

I, George B. Stramback, being duly sworn, depose and state as follows:

(1) I am Project Manager, Regulatory Services, General Electric Company ("GE") and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report NEDC-33085P, *Safety Analysis Report for Edwin I. Hatch Units 1 and 2 Thermal Power Optimization,* Class III (GE Proprietary Information), dated December 2002. The proprietary information is identified by a double underline inside square brackets.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), 2.790(a)(4), and 2.790(d)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information, which fit into the definition of proprietary information, are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals cost or price information, production capacities, budget levels, or commercial strategies of General Electric, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, of potential commercial value to General Electric;

e. Information, which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reason described in items (4)a. and (4)b., above.

(5) The information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GE are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR").

The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GE would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GE of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 13th day of December 2002.

George B. Stramback
George B. Stramback
General Electric Company

Enclosure 1

Request for License Amendment:
Measurement Uncertainty Recapture Power Uprate
10 CFR 50.92 Significant Hazards Evaluation and Environmental Assessment

A. 10 CFR 50.92 Significant Hazards Evaluation

The standards used to arrive at a determination that a request for amendment does not involve a significant hazards consideration are included in 10 CFR 50.92, which states that operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

Description of Proposed Change

Southern Nuclear Company (SNC) is proposing a change to the Edwin I. Hatch Nuclear Plant Unit 1 and Unit 2 Operating Licenses and Technical Specifications associated with an increase in the licensed power level. The change proposes a 1.5 % increase in licensed reactor core thermal power level (an increase in reactor power level from 2763 MWt to 2804 MWt). This change results from the increased accuracy of the feedwater flow measurements utilizing high-accuracy ultrasonic flow measurement instrumentation. This results in a more accurate determination of reactor core thermal power level. The basis for this change is consistent with the July 2000 revision to Appendix K to Title 10 Code of Federal Regulations (CFR) Part 50, which allows operating reactor licensees to use an uncertainty factor of ≤2 % of rated reactor thermal power in analyses of postulated design basis loss-of-coolant accidents.

Bases for No Significant Hazards Determination

SNC's conclusion that the proposed change to the Plant Hatch Unit 1 and Unit 2 Operating Licenses and Technical Specifications does not involve a significant hazards consideration is based upon the following:

1. *The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated*

   The comprehensive analytical efforts performed to support the proposed uprate conditions included a review and evaluation of all components and systems that could be affected by this change. Performance requirements for these systems were evaluated and found acceptable. Furthermore, evaluation of accident analyses confirmed the effects of the proposed uprate are bounded by the current dose analyses. The systems will function as designed. The performance requirements for these systems were evaluated and found acceptable.

The primary loop components (e.g., reactor vessel, reactor internals, control rod drive housings, piping and supports, and recirculation pumps) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, the probability of a structural failure of these components is not increased as a result of this change.

The Nuclear Steam Supply System (NSSS) systems will still perform their intended design functions during normal and accident conditions. The balance-of-plant (BOP) systems and components will continue to meet their applicable structural limits and perform their intended design functions. Thus, the probability of a structural failure of these components is not increased as a result of this change.

The NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves still meet design sizing requirements at the uprated power level.

Because the integrity of the plant will not be affected by operation at the uprated condition, SNC concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions. The reduced uncertainty in the flow input to the core thermal power uncertainty measurement allows most of the current safety analyses to be used, with small changes to the core operating limits, to support operation at a core power of 2804 MWt. Other analyses performed at a nominal power level were either evaluated or reperformed for the 1.5% increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met at the 1.5% uprate conditions. Thus, all Plant Hatch Final Safety Analysis Report accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences were reviewed and determined to bound operation at the 1.5% uprated condition. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.*

   No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed change will have no adverse effect on any safety-related system or component and does not challenge the performance or integrity of any safety-related system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. *The proposed amendment will not involve a significant reduction in a margin of safety.*

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Analyses of the primary fission product barriers confirm that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance with the required acceptance criteria. As appropriate, all evaluations were performed using methods that were either reviewed and approved by the NRC, or are in compliance with regulatory review guidance and standards. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the foregoing analysis and the supporting safety analyses, SNC has concluded that the three standards of 10 CFR 50.92(c) are satisfied, and, therefore, the amendment request involves no significant hazards consideration.

B. Environmental Assessment

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the posed license amendment will not:

(i) involve a significant hazards consideration,

(ii) result in a significant change in the types, or a significant increase in the amounts of any effluent that may be released offsite, or

(iii) result in a significant increase in individual or cumulative occupational exposure.

SNC has determined the proposed licensing amendment meets the eligibility requirements for categorical exclusion as set forth in 10 CFR 51.22. Accordingly, no environmental impact statement associated with the issuance of this amendment need be prepared. The basis for this determination is as follows:

(i) Section A of Enclosure 1 provides the justification as to why this proposed change to the licensed power level does not involve a significant hazards consideration.

(ii) The proposed increase to the maximum licensed power level has been reviewed with respect to offsite releases, with the conclusion that 10 CFR 20 limits continue to be met and that the processing of liquid and gaseous radwaste limits are not adversely affected. Consequently, the proposed change does not involve a significant change in the types of effluents or the amounts of effluents released offsite.

(iii) Operating Plant Hatch at 1.5% above the current maximum power level will not result in an increase to typical occupational exposures. Therefore, an increase in individual or cumulative occupational exposure is not expected.

Enclosure 2

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Bases for Change Request

Plant Hatch Unit 1 and Unit 2 are currently licensed to operate at a maximum rated thermal power (RTP) of 2763 MWt. This power level is supported by a number of analyses and evaluations performed with an RTP uncertainty of ≥ 2%, through either 10 CFR 50, Appendix K, "ECCS Evaluations Models," or Regulatory Guide (RG) 1.49, "Power Levels of Water-Cooled Nuclear Power Plants," May 1973. By applying a reduced thermal power uncertainty to these analyses, Southern Nuclear Operating Company (SNC) can justify increasing Plant Hatch's reactor thermal power level, while remaining within the boundary of these specific analyses. Thus, SNC requests approval to increase the licensed RTP of Plant Hatch Unit 1 and Unit 2 by 1.5% to 2804 MWt. This increase in power level will be accomplished by using a more accurate main feedwater flow measurement system that will provide input into the calculated core thermal power of each unit.

The proposed changes to the Operating Licenses and Technical Specifications are based upon safety analyses performed per NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 Thermal Power Optimization"(1) which provides the following:

- An evaluation of the effects of increasing power level by 1.5% due to an increase in the accuracy of the feedwater flow measurement.
- A summary of the results of all significant safety evaluations that justify increasing the licensed thermal power of Plant Hatch Unit 1 and Unit 2 to 2804 MWt.

The overall approach to the 1.5% power increase is consistent with the methodology established in NEDC-32938-P, "Licensing Topical Report, Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization."(2) NEDC-32938P addresses power increases up to 1.5% of current licensed thermal power (CLTP), producing up to an approximate 2% increase in steam flow to the turbine-generator. A higher steam flow is achieved by increasing the reactor power along the current rod and core flow control lines. The increase in CLTP requires that a limited number of operating parameters be changed, some setpoints adjusted, and some instruments recalibrated. Plant procedures must be revised to reflect the power level increase, and tests similar to some of the original startup tests will be performed.

PROPOSED CHANGES

Each Operating License (OL) and Technical Specifications (TS) change, as well as the justification for each change, is provided below. Unless noted otherwise, the proposed changes are identical for Unit 1 and Unit 2.

Renewed Facility Operating Licenses DPR-57 (Unit 1) and NPF-5 (Unit 2)

Page 4, Section C.(1), Maximum Power Level

This proposed revision changes the maximum power level from "2,763 megawatts thermal" to "2,804 megawatts thermal." The OLs are revised to read as follows:

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not to exceed 2,804 megawatts thermal.

The following evaluations were conducted, as summarized in NEDC-33085P(1), in accordance with the criteria of Appendix B to NEDC-32938-P.(2)

All safety aspects of the plant affected by a 1.5% increase in the thermal power level, including the nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems, were evaluated.

Evaluations and reviews were based upon licensing criteria, codes, and standards applicable to Plant Hatch. The only change to the previously established licensing basis for the plant is the increase in power level.

Evaluations and/or analyses were performed using NRC-approved analysis methods for the Final Safety Analysis Report (FSAR) accidents, transients, and special events affected by thermal power optimization (TPO).

Evaluations and reviews of the NSSS systems and components, containment structures, and BOP systems and components show continued compliance to the codes and standards applicable to the current plant licensing basis; i.e., no change to comply with more recent codes and standards is proposed due to TPO.

NSSS systems and components were reviewed to confirm they continue to comply with the functional and regulatory requirements specified in the FSAR and/or applicable reload license.

All plant systems and components affected by an increase in thermal power level were reviewed to ensure there is no significant increase in challenges to the safety systems.

Evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, environmental issues, design basis accidents, and previous licensing evaluations were performed within NEDC-33085P.(1) Therefore, NEDC-33085P demonstrates that Plant Hatch can safely operate at a power level of 2804 MWt.

TS 1.0 USE AND APPLICATION

TS 1.1 Definitions

RATED THERMAL POWER (RTP) RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2804 MWt.

The value of the definition for RTP is changed from 2763 MWt to 2804 MWt. For the basis of this change, refer to the basis of the change to the maximum power level for the OLs above.

TS 2.0 SAFETY LIMITS (SLs)

General Design Criterion (GDC) 10 requires, and SLs ensure, specified acceptable fuel design limits are not exceeded during steady-state operation, normal operational transients, and anticipated operational occurrences (AOOs).

TS 2.1.1 Reactor Core SLs

Fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses that occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that produce the onset of transition boiling [i.e., Minimum Critical Power Ratio (MCPR) = 1.00]. These conditions represent a significant departure from the condition intended by design for planned operation. The MCPR fuel cladding integrity SL ensures that during normal operation and AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Operation above the boundary of the nucleate boiling regime can result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form that may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

TS 2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow: THERMAL POWER shall be ≤ 24% Rated THERMAL POWER (RTP).

The thermal power value for this SL is being reduced from ≤ 25% RTP to ≤ 24% RTP. The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling. Operating thermal limits ensure regulatory limits and/or SLs are not exceeded for a range of postulated events (e.g., transients, loss of coolant accidents (LOCAs)). Cycle-specific core configurations, which are evaluated for each reload, confirm TPO RTP capability and establish or confirm cycle-specific limits.

The historical 25% of RTP value [for the TS SL, some thermal limits monitoring Limiting Condition for Operation (LCO) thresholds, and some Surveillance Requirement (SR) thresholds] is based upon generic analyses (evaluated up to ~ 50% of original RTP) applicable to the plant with the highest average bundle power (at 100% RTP) for any product line. As originally licensed, the highest average bundle power (at 100% RTP) for any BWR-6 is 4.8 MWt/bundle. The historical 25% RTP value is conservative. However, this percent value should be reduced when any plant is uprated such that at 100% of uprated power, the average bundle power is greater than the original generic basis of 4.8 MWt/bundle. Therefore, to maintain the same basis with respect to absolute thermal power, if the uprated average bundle power is > 4.8 MWt/bundle, the percent RTP value is revised to equal (25% x 4.8 MWt/bundle x no. of bundles ÷ total uprated MWt). For the Plant Hatch TPO uprate, the average bundle power is > 4.8 MWt/bundle. Therefore, the SL percent RTP basis, some thermal limits monitoring LCOs, and some SR percent RTP thresholds are reduced from 25% RTP to 24% RTP.

TS 3.2 POWER DISTRIBUTION LIMITS
TS 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

The 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. APLHGR limits are specified to ensure certain fuel design limits are not exceeded during AOOs and the peak cladding temperature (PCT) during the postulated design basis LOCA does not exceed the limits specified in 10 CFR 50.46.

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.
APPLICABILITY: THERMAL POWER ≥ 24% RTP.
The Applicability for this LCO is being changed from "THERMAL POWER ≥ 25% RTP" to "THERMAL POWER ≥ 24% RTP" to coincide with the change made to the SL in TS 2.1.1. (For the details of the justification for this change, see the justification for TS 2.1.1.1 above.)

The APLHGR limits are primarily derived from fuel design evaluations, and LOCA and transient analyses assumed to occur at high power levels. Design calculations and operating experience show that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at thermal power levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

REQUIRED ACTION B.1 Reduce THERMAL POWER TO < 24% RTP.
Required Action B.1 is being changed from "Reduce THERMAL POWER to < 25% RTP" to "Reduce THERMAL POWER to < 24% RTP" to coincide with the Applicability for this LCO.

SR 3.2.1.1 FREQUENCY Once with 12 hours after ≥ 24% RTP AND 24 hours thereafter
The Frequency for SR 3.2.1.1 is being changed from "Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter" to "Once within 12 hours after ≥ 24% RTP AND 24 hours thereafter" to correspond with the change to the Applicability.

TS 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

The MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR SL is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated. The operating limit (OL) MCPR is established to ensure no fuel damage will result during AOOs. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition, the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.
APPLICABILITY: THERMAL POWER ≥ 24% RTP.
The Applicability for this LCO is being changed from "THERMAL POWER ≥ 25% RTP" to "THERMAL POWER ≥ 24% RTP" to coincide with the change made to the SL in TS 2.1.1. (For the details of the justification for this change, see the justification for TS 2.1.1.1 above.) The MCPR OLs are primarily derived from transient analyses assumed to occur at high power levels. Below 24% RTP, the reactor is operating at a minimum recirculation pump speed, and the moderator void ratio is small. Surveillance of thermal limits below 24% RTP is unnecessary due to the large inherent margin that ensures the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 24% RTP is > 3.5. Studies of the variation of limiting transient behavior were performed over the range of power and flow conditions. These studies encompassed the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin between performance and MCPR requirements is expected, and that margins increase as power is reduced to 24% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at thermal power levels < 24% RTP, the reactor is operating with substantial margin to the MCPR limits; thus, this LCO is not required.

REQUIRED ACTION B.1 Reduce THERMAL POWER TO < 24% RTP.
Required Action B.1 is being changed from "Reduce THERMAL POWER to < 25% RTP" to "Reduce THERMAL POWER to < 24% RTP" to coincide with the Applicability for this LCO.

SR 3.2.2.1 FREQUENCY Once with 12 hours after ≥ 24% RTP AND 24 hours thereafter
The Frequency for SR 3.2.2.1 is being changed from "Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter" to "Once within 12 hours after ≥ 24% RTP AND 24 hours thereafter" to correspond with the change to the Applicability.

TS 3.3 INSTRUMENTATION
TS 3.3.1.1 Reactor Protection System (RPS) Instrumentation

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, thereby preserving the integrity of the fuel cladding and the reactor coolant system (RCS) and minimizing the energy that must be absorbed following a LOCA. This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as LCOs on other reactor system parameters and equipment performance. The LSSS are defined in the Technical Specifications as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

Table 3.3.1.1-1 Reactor Protection System Instrumentation
Function 2.b APRM - Simulated Thermal Power - High

This Function monitors neutron flux to approximate the thermal power being transferred to the reactor coolant. The average power range monitor (APRM) neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the thermal power in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the APRM Neutron Flux - High Function Allowable Value.

The APRM Simulated Thermal Power - High Function provides protection against transients where thermal power increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring the MCPR SL is not exceeded. During these events, the thermal power increase does not significantly lag the neutron flux response, and because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, thermal power lags the neutron flux, and the APRM Neutron Flux - High Function will provide a scram signal before the APRM Simulated Thermal Power - High Function setpoint and associated time delay are exceeded.

SR 3.3.1.1.2 Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 24% RTP. [NOTE - Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP.]

The condition for this SR is being changed from "while operating at ≥ 25% RTP" to "while operating at ≥ 24% RTP." The associated Note is also being changed to show that this SR is "Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP." Both of these changes are being made to coincide with the change made to the SL specified in TS 2.1.1. (For the details of the justification for this change, see the justification for TS 2.1.1.1 above.)

To ensure the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based upon minor changes in local power range monitor (LPRM) sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 24% RTP requires the SR to be met at only ≥ 24% RTP, because it is difficult to accurately maintain APRM indication of core thermal power consistent with a heat balance when operating at < 24% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 24% RTP, the Surveillance must have been performed satisfactorily within the last 7 days, in accordance with SR 3.0.2. The Note allows an increase in THERMAL POWER above 24% if the 7-day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 24% RTP. Twelve hours is based upon operating experience and a reasonable time in which to complete the SR.

Allowable Value: **≤ 0.57 W + 56.8% RTP and ≤ 115.5% RTP[(b)]**
[(b) 0.57 W + 56.8% - 0.57 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."]

The Allowable Value for this TS Function is being changed from "≤ 0.58W + 58% RTP and ≤ 115.5% RTP" to "≤ 0.57W + 56.8% RTP and ≤ 115.5% RTP." The footnote is being changed from "(b) 0.58W + 58% - 0.58 ΔW RTP when reset for single loop operation per LCO 3.4.1, 'Recirculation Loops Operating'" to "(b) 0.57 W + 56.8% - 0.57 ΔW RTP when reset for single loop operation per LCO 3.4.1, 'Recirculation Loops Operating'." The flow-referenced APRM Allowable Values for both two-loop operation and single-loop operation are unchanged in units of absolute core thermal power versus recirculation drive flow. Because the Allowable Values are expressed in percent of RTP, they decrease in proportion to the power uprate. This is the same approach that is taken for generic BWR uprates in NEDC-32424P-A.[(3)] No significant effects on the instrument errors or uncertainties will result from this uprate. Therefore, the nominal trip setpoints (NTSPs) are established by directly incorporating the change in the Allowable Values.

Table 3.3.1.1-1 **Reactor Protection System Instrumentation**
Function 2.c APRM - Neutron Flux - High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the FSAR overpressurization protection analysis, the APRM Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the safety relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis takes credit for the APRM Neutron Flux - High Function to terminate the CRDA.

SR 3.3.1.1.2 **Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 24% RTP. [NOTE- Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP.]**

The condition for this SR is being changed from "while operating at ≥ 25% RTP" to "while operating at ≥ 24% RTP." The associated Note is also being changed to show that this SR is

"Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP." Both of these changes are being made to coincide with the change made to the SL in TS 2.1.1. (For the details of the justification for this change, see the justification for TS 2.1.1.1 above.)

To ensure the APRMs are accurately indicating the true core average power, they are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based upon minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 24% RTP requires the SR to be met at only ≥ 24% RTP, because it is difficult to accurately maintain APRM indication of core thermal power consistent with a heat balance when < 24% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 24% RTP, the Surveillance must have been performed satisfactorily within the last 7 days, in accordance with SR 3.0.2. The Note allows an increase in thermal power above 24% if the 7-day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 24% RTP. Twelve hours is based upon operating experience and a reasonable time in which to complete the SR.

Table 3.3.1.1-1 Reactor Protection System Instrumentation
Function 8 Turbine Stop Valve - Closure

Closure of the turbine stop valves (TSVs) results in the loss of a heat sink, which produces RPV pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that will result from the closure of these valves. The TSV - Closure Function is the primary scram signal for the turbine trip event analyzed in the FSAR.

Per NEDC-33085P[1], the TSV closure scram, TCV fast closure scram, and end-of-cycle recirculation pump trip (EOC-RPT) bypasses allow this scram and EOC-RPT to be bypassed, when reactor power is sufficiently low, such that the scram and EOC-RPT functions are not needed to mitigate a turbine-generator trip. This power level is used to determine the actual trip setpoint, which comes from the turbine first-stage pressure (TFSP) switches. The TFSP setpoint is chosen to allow operational margin so that scrams and RPTs can be avoided by transferring steam to the turbine bypass system during turbine-generator trips at low power.

Table 3.3.1.1-1 Function 8 Applicable Modes or Other Specified Conditions
SR 3.3.1.1.11 Verify Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.

The condition for this SR is being changed from "≥ 28% RTP" to "≥ 27.6% RTP." (The value changes in the SR, as well as Table 3.3.1.1-1, Function 8, under the Applicable Modes or Other Specified Conditions column 8.) Per Section F.4.2.3 of NEDC-32938-P[2], the TSV closure scram, the TCV fast closure scram, and EOC-RPT bypass value in percent RTP is reduced by the ratio of the power increase. The new value does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant startup is unchanged from the currently licensed thermal power maneuvering range. Also, since there is no change in terms of absolute power, actual pressure switch settings are unchanged. Therefore, no physical changes are required to implement this change.

REQUIRED ACTION E.1 Reduce THERMAL POWER to < 27.6% RTP.
Required Action E.1 is being changed from "Reduce THERMAL POWER to < 28% RTP" to "Reduce THERMAL POWER to < 27.6% RTP" to coincide with the change in the value in SR 3.3.1.1.11.

Table 3.3.1.1-1 Reactor Protection System Instrumentation
Function 9 Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs results in the loss of a heat sink that produces RPV pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that result from the closure of these valves. The TCV Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in the FSAR. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT system, ensures the MCPR SL is not exceeded. TCV Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve.

Per NEDC-33085P[1], the TSV closure scram, TCV fast closure scram, and EOC-RPT bypasses allow this scram and EOC-RPT to be bypassed when reactor power is sufficiently low, such that the scram and EOC-RPT Functions are not needed to mitigate a turbine-generator trip. This power level is used to determine the actual trip setpoint that comes from the TFSP switches. The TFSP setpoint is chosen to allow operational margin so that scrams and RPTs can be avoided, by transferring steam to the turbine bypass system during turbine-generator trips at low power.

SR 3.3.1.1.11 Verify Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.
The condition for this SR is being changed from "≥ 28% RTP" to "≥ 27.6% RTP." (The value changes in the SR, as well as Table 3.3.1.1-1, Function 9, under the Applicable Modes or Other Specified Conditions column.) Per Section F.4.2.3 of NEDC-32938-P[2], the TSV closure scram, the TCV fast closure scram, and EOC-RPT bypass value in percent RTP is reduced by the ratio of the power increase. The new value does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant startup is unchanged from the currently licensed thermal power maneuvering range. Also, since there is no change in terms of absolute power, actual pressure switch settings are unchanged. Therefore, no physical changes are required to implement this change.

REQUIRED ACTION E.1 Reduce THERMAL POWER to < 27.6% RTP.
Required Action E.1 is being changed from "Reduce THERMAL POWER to < 28% RTP" to "Reduce THERMAL POWER to < 27.6% RTP" to coincide with the change in the value in SR 3.3.1.1.11.

TS 3.3.2.2 Feedwater and Main Turbine Trip High Water Level Instrumentation

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. With excessive feedwater flow, the water level in the RPV rises toward the high water level setpoint, causing the trip of the two feedwater pump turbines and the main turbine. A trip of the feedwater pump turbines limits further increase in RPV water level by limiting further addition of feedwater to the RPV. A trip of the main turbine and closure of the stop valves protect the turbine from damage due to water entering the turbine. The feedwater and main turbine high-water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event. The high-level trip indirectly initiates a reactor scram from the main turbine trip and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

LCO 3.3.2.2 Three channels of feedwater and main turbine trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.
The Applicability for this LCO is being changed form "THERMAL POWER ≥ 25% RTP" to "THERMAL POWER ≥ 24% RTP" to coincide with the change made to the SL in TS 2.1.1 (For the details of the justification for this change, see the justification for TS 2.1.1.)

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 24% RTP to ensure that the specified acceptable fuel design limits are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1 "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 24% RTP; therefore, these requirements are only necessary when operating at or above this power level.

REQUIRED ACTION C.1 Reduce THERMAL POWER to < 24% RTP.
Required Action C.1 is being changed form "Reduce THERMAL POWER to <25% RTP" to "Reduce THERMAL POWER TO <24% RTP" to coincide with the Applicability for this LCO.

TS 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

The EOC-RPT instrumentation initiates an RPT to reduce the peak RPV pressure and power resulting from the turbine trip or generator load rejection transients, thus providing additional margin to core thermal MCPR SLs. The need for the additional negative reactivity in excess of that normally inserted on a scram reflects end-of-cycle reactivity considerations. Depending upon the MCPR OL, flux shapes at the end of cycle can be such that the control rods will not be able to ensure thermal limits are maintained by inserting sufficient negative reactivity during the first few feet of control rod travel upon a scram caused by either TSV - Closure or TCV Fast Closure, Trip Oil Pressure - Low. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity at a faster rate than the control rods can add negative reactivity. The EOC-RPT allows a margin improvement, which in turn allows a reduction in the MCPR OL.

The EOC-RPT instrumentation is composed of sensors that detect initiation of closure of the TSVs or fast closure of the TCVs, combined with relays, logic circuits, and fast-acting circuit breakers that interrupt power from the recirculation pump motor generator set to each of the recirculation pump motors. The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an EOC-RPT signal to the trip logic. When the RPT breakers trip open, the recirculation pumps coast down under their own inertia. The EOC-RPT has two identical trip systems, either of which can actuate an RPT. The EOC-RPT initiation logic is designed to be single-failure proof and, therefore, is highly reliable.

Per NEDC-33085P[(1)], the TSV closure scram, TCV fast closure scram, and EOC-RPT bypasses allow this scram and EOC-RPT to be bypassed when reactor power is sufficiently low, such that the scram and EOC-RPT functions are not needed to mitigate a turbine-generator trip. This power level is used to determine the actual trip setpoint, which comes from the TFSP switches. The TFSP setpoint is chosen to allow operational margin so that scrams and RPTs can be avoided by transferring steam to the turbine bypass system during turbine-generator trips at low power.

APPLICABILITY: THERMAL POWER ≥ 27.6% RTP.
The Applicability for this LCO is being changed from "THERMAL POWER ≥ 28% RTP" to "THERMAL POWER ≥ 27.6% RTP." Per Section F.4.2.3 of NEDC-32938-P[(2)], the TSV closure scram, the TCV fast closure scram, and EOC-RPT bypass value in percent RTP is reduced by the ratio of the power increase. The new value does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant startup is unchanged from the currently licensed thermal power maneuvering range. Also, since there is no change in terms of absolute power, actual pressure switch settings are unchanged. Therefore, no physical changes are required to implement this change.

REQUIRED ACTION C.2 Reduce THERMAL POWER to < 27.6% RTP.
Required Action C.2 is being changed from "Reduce THERMAL POWER to < 28% RTP" to "Reduce THERMAL POWER to < 27.6% RTP" to coincide with the change to the applicability for the LCO.

LCO 3.3.4.1.a.1 EOC-RPT Instrumentation Function – TSV - Closure

SR 3.3.4.1.2 Verify TSV – Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.
The condition for this SR is being changed from "≥ 28% RTP" to "≥ 27.6% RTP." Per Reference 2, Section F.4.2.3, the TSV closure scram, the TCV fast closure scram, and EOC-RPT bypass value in percent RTP is reduced by the ratio of the power increase. The new value does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant

startup is unchanged from the currently licensed thermal power maneuvering range. Also, since there is no change in terms of absolute power, there is no change in the actual pressure switch settings. Therefore, no physical changes are required to implement this change.

LCO 3.3.4.1.a.2 EOC-RPT Instrumentation Function -- TCV Fast Closure, Trip Oil Pressure - Low

SR 3.3.4.1.2 Verify TSV – Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.

The condition for this SR is being changed from "≥ 28% RTP" to "≥ 27.6% RTP." Per Section F.4.2.3 of NEDC-32938-P[2], the TSV closure scram, the TCV fast closure scram, and EOC-RPT bypass value in percent RTP is reduced by the ratio of the power increase. The new value does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. The maneuvering range for plant startup is unchanged from the currently licensed thermal power maneuvering range. Also, since there is no change in terms of absolute power, actual pressure switch settings are unchanged. Therefore, no physical changes are required to implement this change.

TS 3.7 PLANT SYSTEMS
TS 3.7.7 Main Turbine Bypass System

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is approximately 21% of the turbine design steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of three valves connected to the MSLs between the main steam isolation valves and the turbine stop valves. Hydraulic cylinders operate each of these three valves. The bypass valves are controlled by the pressure regulation function of the Turbine Electrohydraulic Control System. The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

The Main Turbine Bypass System is assumed to function during the feedwater controller failure to maximum flow demand. Opening the bypass valves during the pressurization event (subsequent to the resulting main turbine trip) mitigates the increase in RPV pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)" limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable

APPLICABILITY: THERMAL POWER ≥ 24% RTP.
The Applicability for this LCO is being changed form "THERMAL POWER ≥ 25% RTP" to "THERMAL POWER ≥ 24% RTP" to coincide with the change made to the SL in TS 2.1.1 (For the details of the justification for this change, see the justification for TS 2.1.1.)

The Main Turbine Bypass System is required to be OPERABLE at ≥ 24% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit arte not violated during the feedwater controller failure to maximum flow demand transient. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR), " and LCO 3.2.2, sufficient margin to these limits exists at < 24% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

REQUIRED ACTION B.1 Reduce THERMAL POWER TO < 24% RTP.
Required Action B.1 is being changed from "Reduce THERMAL POWER TO < 25% RTP" to "Reduce THERMAL POWER TO < 24% RTP" to coincide with the change to the applicability for the LCO.

SUMMARY OF REQUIRED CHANGES TO SAFETY/EMERGENCY SYSTEM SETTINGS

The following is a summary of the required changes to safety system settings and scaling to accomplish the objectives of the Appendix K Measurement Uncertainty Recapture Power Uprate Project.

1. Scaling for the APRMs will require change such that 0 -100% RTP corresponds to 0 to 2804 MWt.

2. The setting for the Flow-Referenced Simulated Thermal Power High Scram (Function 2.b of TS Table 3.3.1.1-1) will be changed to reflect a setpoint function of 0.57 W + 56.8% RTP for two-loop operation and 0.57 W + 56.8% - 0.57 ΔW RTP for single-loop operation.

Notes:

A. Since the actual power setting for the bypass for the TSV and TCV Closure scram and EOC-RPT instrumentation does not change, the pressure switches that provide this function will not require rescaling.

B. Since the functions that occur in terms of percent RTP act on a percentage of the APRM signal, and since the fixed settings did not change in terms of percent RTP, the settings do not require change. Only the range of the APRM requires rescaling in this instance, per item 1 above.

C. Other changes to the plant systems not covered in the TS will be performed per the 10 CFR 50.59 evaluation process as a part of the implementation of this project.

REFERENCES

1. NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," December, 2002.

2. NEDC-32938-P, "Licensing Topical Report, Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," July 2000.

3. NEDC-32424P-A, "Licensing Topical Report, Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," February 1995.

Enclosure 3

Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Page Change Instructions

Unit 1 Operating License No. DPR-57

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Unit 1 Technical Specifications

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Unit 2 Operating License No. NPF-5

Page	Instruction
4	Replace

Unit 2 Technical Specifications

Page	Instruction
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2.0-1	Replace
3.2-1	Replace
3.2-2	Replace
3.3-2	Replace
3.3-3	Replace
3.3-5	Replace
3.3-7	Replace
3.3-9	Replace

Enclosure 3

Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Page Change Instructions

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1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.6, Startup and Power Test Program, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2804 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager, the corporate executive responsible for overall plant nuclear safety, and the offsite review committee.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 27.6% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations. AND I.2 Restore required channels to OPERABLE.	12 hours 120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Be in MODE 2.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP. ---------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 24% RTP.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.	24 months
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	---NOTES--- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	---NOTE--- Neutron detectors are excluded. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux - High	2	2(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	2(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	2(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	2(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux - High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 0.57W + 56.8% RTP and ≤ 115.5% RTP(b)
c. Neutron Flux - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1, 2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.57W + 56.8% - 0.57 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8.	Turbine Stop Valve - Closure	≥ 27.6% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 10% closed
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 27.6% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 600 psig
10.	Reactor Mode Switch - Shutdown Position	1, 2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
		5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11.	Manual Scram	1, 2	1	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
		5(a)	1	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine Trip High Water Level Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 27.6% RTP.

ACTIONS

------------------------------NOTE------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	OR	
	A.2 -----------NOTE----------- Not applicable if inoperable channel is the result of an inoperable breaker. ------------------------------ Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 Restore EOC-RPT trip capability. OR B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours 2 hours
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump from service. OR C.2 Reduce THERMAL POWER to < 27.6% RTP.	4 hours 4 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days on an ALTERNATE TEST BASIS
SR 3.3.4.1.2	Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.	24 months

(continued)

3.7 PLANT SYSTEMS

3.7.7 Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.7.2	Perform a system functional test.	24 months
SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2804 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency; so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager, the corporate executive responsible for overall plant nuclear safety, and the offsite review committee.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP AND 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 27.6% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations.	12 hours
	AND	
	I.2 Restore required channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Be in MODE 2.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 24% RTP. ----- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 24% RTP.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.	24 months
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	-------------------------NOTES------------------------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	-------------------------NOTES------------------------- 1. Neutron detectors are excluded. 2. (Not used.) 3. For Function 5, "*n*" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux - High	2	2(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	2(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	2(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	2(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux - High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 0.57W + 56.8% RTP and ≤ 115.5% RTP(b)
c. Neutron Flux - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1, 2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) 0.57W + 56.8% - 0.57 ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Turbine Stop Valve - Closure	≥ 27.6% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 27.6% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 600 psig
10. Reactor Mode Switch - Shutdown Position	1, 2	2	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5[(a)]	2	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11. Manual Scram	1, 2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5[(a)]	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine Trip High Water Level Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ 27.6% RTP.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	OR	
	A.2 -----------NOTE---------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------------- Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 Restore EOC-RPT trip capability. OR B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours 2 hours
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump from service. OR C.2 Reduce THERMAL POWER to < 27.6% RTP.	4 hours 4 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days on an ALTERNATE TEST BASIS
SR 3.3.4.1.2	Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 27.6% RTP.	24 months

(continued)

3.7 PLANT SYSTEMS

3.7.7 Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.7.2	Perform a system functional test.	24 months
SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

Enclosure 3

Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Page Change Instructions

Unit 1 Bases

Page	Instruction
B 2.0-2	Replace
B 3.2-1	Replace
B 3.2-3	Replace
B 3.2-4	Replace
B 3.2-6	Replace
B 3.2-7	Replace
B 3.3-3	Replace
B 3.3-7	Replace
B 3.3-16	Replace
B 3.3-17	Replace
B 3.3-24	Replace
B 3.3-28	Replace
B 3.3-53	Replace
B 3.3-54	Replace
B 3.3-56	Replace
B 3.3-76	Replace
B 3.3-77	Replace
B 3.3-78	Replace
B 3.3-80	Replace
B 3.3-81	Replace
B 3.3-141	Replace
B 3.4-10	Replace
B 3.7-31	Replace
B 3.7-35	Replace
B 3.7-36	Replace

Unit 2 Bases

Page	Instruction
B 2.0-2	Replace
B 3.2-1	Replace
B 3.2-3	Replace
B 3.2-4	Replace
B 3.2-6	Replace
B 3.2-7	Replace

Enclosure 3

Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Page Change Instructions

Unit 2 Bases (Continued)

Page	Instruction
B 3.3-3	Replace
B 3.3-7	Replace
B 3.3-16	Replace
B 3.3-17	Replace
B 3.3-24	Replace
B 3.3-28	Replace
B 3.3-53	Replace
B 3.3.54	Replace
B 3.3.56	Replace
B 3.3-76	Replace
B 3.3-77	Replace
B 3.3-78	Replace
B 3.3-80	Replace
B 3.3-81	Replace
B 3.3-141	Replace
B 3.4-10	Replace
B 3.7-31	Replace
B 3.7-35	Replace
B 3.7-36	Replace

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints [LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"], in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR SL.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 24% RTP for reactor pressure < 785 psig is conservative.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 5, 6, and 7). Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 24% RTP and the previously mentioned bypass power level.

(continued)

BASES (continued)

APPLICABILITY	The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.
ACTIONS	<u>A.1</u> If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification. <u>B.1</u> If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to in a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	<u>SR 3.2.1.1</u> APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. FSAR, Chapter 3.
3. FSAR, Chapter 6.
4. FSAR, Chapter 14.
5. NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," August 1989.
6. NEDO-24395, "Load Line Limit Analysis," October 1980.
7. NEDC-30474-P "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E.I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.
11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
12. NEDC-33085-P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 Thermal Power Optimization," November 2002.

BASES

APPLICABLE SAFETY ANALYSES (continued)

benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 24% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 24% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 24% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 24% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 24% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 24% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions specified in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals. The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses ≥ 27.6% RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High (Setdown) (continued)

abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 24% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 24% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 24% RTP.

The Average Power Range Monitor Neutron Flux - High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power - High

The Average Power Range Monitor Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux - High Function Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if the TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. This Function is not required when THERMAL POWER is < 27.6% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. This Function is not required when THERMAL POWER is < 27.6% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with two channels, each of which provides input into one of the RPS manual scram logic channels.

(continued)

BASES

SURVEILLANCE REUIREMENTS

SR 3.3.1.1.1 (continued)

between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 24% RTP is provided that requires the SR to be met only at ≥ 24% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 24% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 24% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 24% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 24% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

POWER is ≥ 27.6% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 27.6% RTP to ensure that the calibration is valid.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 27.6% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 18.

SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that 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. For MSIV - Closure, SDV Water Level - High (Float Switch), and TSV - Closure Functions, this SR also includes a physical inspection and actuation of the switches. For the APRM Simulated Thermal Power - High Function, this SR also includes calibrating the associated recirculation loop flow channel.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the IRM SRs

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B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level setpoint, causing the trip of the two feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level - High instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high level trip indirectly initiates a reactor scram from the main turbine trip (above 27.6% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The LCO requires three channels of the Reactor Vessel Water Level - High instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.2. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions (nominal trip setpoint). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 24% RTP to ensure that the specified acceptable fuel design limits are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 24% RTP; therefore, these requirements are only necessary when operating at or above this power level.

(continued)

BASES

ACTIONS

B.1 (continued)

not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 24% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 24% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 24% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2 and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 27.6% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

Turbine Stop Valve - Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink and increases reactor pressure, neutron flux, and heat flux that must be limited. Therefore, an RPT is initiated on a TSV - Closure signal before the TSVs are completely closed in anticipation of the effects that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each valve. While there are two separate position switches associated with each stop valve, only the signal from one switch for each TSV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

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BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. Below 27.6% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 27.6% RTP. Below 27.6% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Neutron Flux - High Functions of the RPS are adequate to maintain the necessary margin to the MCPR SL.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 27.6% RTP within 4 hours. Alternately, the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 27.6% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day on an ALTERNATE TEST BASIS Frequency is based on a review of the surveillance test history and Reference 8.

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 27.6% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 27.6% RTP to ensure that the calibration is valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 27.6% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met with the channel considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 7.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. 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. For the TSV - Closure Function, this SR also includes a physical inspection and actuation of the switches.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 10 of 11 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section 9.4 and Appendix E (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2804 MWt.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits [LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"] may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 24% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure to maximum flow demand transient. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < 24% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to

(continued)

BASES

ACTIONS

B.1 (continued)

< 24% RTP. As discussed in the Applicability section, operation at < 24% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection transient. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on a review of the surveillance test history and Reference 5.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in Technical Requirements Manual (Ref. 3). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints [LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"], in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 24% RTP for reactor pressure < 785 psig is conservative.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 5, 6, and 7). Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 24% RTP and the previously mentioned bypass power level.

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 24% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to in a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. FSAR, Chapter 4.
3. FSAR, Chapter 6.
4. FSAR, Chapter 15.
5. NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," August 1989.
6. NEDO-24395, "Load Line Limit Analysis," October 1980.
7. NEDC-30474-P "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E.I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.
11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
12. NEDC-33085-P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 Thermal Power Optimization," November 2002.

BASES

APPLICABLE SAFETY ANALYSES (continued)

benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 24% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 24% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 24% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 24% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 24% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 24% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 24% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 24% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 24% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 24% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50 49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions specified in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals. The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses ≥ 27.6% RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High (Setdown) (continued)

abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 24% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 24% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 24% RTP.

The Average Power Range Monitor Neutron Flux - High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power - High

The Average Power Range Monitor Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux - High Function Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if the TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. This Function is not required when THERMAL POWER is < 27.6% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. This Function is not required when THERMAL POWER is < 27.6% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, to each of the four RPS logic channels, which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

(continued)

BASES

SURVILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 24% RTP is provided that requires the SR to be met only at ≥ 24% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 24% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 24% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 24% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 24% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

POWER is ≥ 27.6% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 27.6% RTP to ensure that the calibration is valid.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 27.6% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 20.

SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that 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. For MSIV - Closure, SDV Water Level - High (Float Switch), and TSV - Closure Functions, this SR also includes a physical inspection and actuation of the switches. For the APRM Simulated Thermal Power - High Function, this SR also includes calibrating the associated recirculation loop flow channel.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the IRM SRs

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND	The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow. With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level setpoint, causing the trip of the two feedwater pump turbines and the main turbine. Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level - High instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic. A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.
APPLICABLE SAFETY ANALYSES	The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high level trip indirectly initiates a reactor scram from the main turbine trip (above 27.6% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR. Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The LCO requires three channels of the Reactor Vessel Water Level - High instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.2. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions (nominal trip setpoint). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 24% RTP to ensure that the specified acceptable fuel design limits are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 24% RTP; therefore, these requirements are only necessary when operating at or above this power level.

(continued)

BASES

ACTIONS

B.1 (continued)

not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 24% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 24% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 24% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2 and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 27.6% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nomial setspoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

<u>Turbine Stop Valve - Closure</u>

Closure of the TSVs and a main turbine trip result in the loss of a heat sink and increases reactor pressure, neutron flux, and heat flux that must be limited. Therefore, an RPT is initiated on a TSV - Closure signal before the TSVs are completely closed in anticipation of the effects that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each valve. While there are two separate position switches associated with each stop valve, only the signal from one switch for each TSV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 27.6% RTP. Below 27.6% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR Safety Limit.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 27.6% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 27.6% RTP. Below 27.6% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Neutron Flux - High Functions of the RPS are adequate to maintain the necessary margin to the MCPR SL.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 27.6% RTP within 4 hours. Alternately, the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 27.6% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

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BASES

SUREVILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day on an ALTERNATE TEST BASIS Frequency is based on a review of the surveillance test history and Reference 8.

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 27.6% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 27.6% RTP to ensure that the calibration is valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 27.6% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met with the channel considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 7.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. 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. For the TSV - Closure Function, this SR also includes a physical inspection and actuation of the switches.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

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B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND	The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."
APPLICABLE SAFETY ANALYSIS	The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 10 of 11 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

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B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Sections 11.3 and 15.1.35 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2804 MWt.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits [LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"] may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 24% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure to maximum flow demand transient. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < 24% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to

(continued)

BASES

ACTIONS

B.1 (continued)

< 24% RTP. As discussed in the Applicability section, operation at < 24% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection transient. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on a review of the surveillance test history and Reference 5.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in Technical Requirements Manual (Ref. 3). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

Enclosure 4

Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

Marked-Up Pages for Operating Licenses, Technical Specifications, and Bases for sample analysis or instrument calibration, or associated with radioactive apparatus or components

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not to exceed 2,763 megawatts thermal.

2,804

(2) <u>Technical Specifications</u>

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 225, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) <u>Fire Protection</u>

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, which was originally submitted by letter dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes

1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR)

The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

MODE

A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Section 13.6, Startup and Power Test Program, of the FSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~2763~~ MWt.
2804

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

24%

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager, the corporate executive responsible for overall plant nuclear safety, and the offsite review committee.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

24%

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP. 24%	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP ≥ 24% <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

24%

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP. 24%	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 25% RTP 24% <u>AND</u> 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~28%~~ RTP. 27.6 %	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations.	12 hours
	AND	
	I.2 Restore required channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Be in MODE 2.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. 24% ----- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP. 24%	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 28% RTP. 27.6%	24 months
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	----------NOTE---------- Neutron detectors are excluded. ---------- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux - High	2	2(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	2(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	2(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	2(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux - High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	0.57W + 56.8% ≤ ~~0.58 W + 58%~~ RTP and ≤ 115.5% RTP(b)
c. Neutron Flux - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1, 2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 58% − 0.58~~ ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

0.57W + 56.8% - 0.57

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8.	Turbine Stop Valve - Closure	≥ ~~28%~~ RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 10% closed
	27.6%					
9.	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ ~~28%~~ RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 600 psig
10.	Reactor Mode Switch - Shutdown Position	1, 2	1	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
		5(a)	1	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11.	Manual Scram	1, 2	1	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
		5(a)	1	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine Trip High Water Level Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ RTP.

24%

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < ~~25%~~ RTP.	4 hours

24%

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~28%~~ RTP.
27.6%

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	OR	
	A.2 -----NOTE----- Not applicable if inoperable channel is the result of an inoperable breaker. ----- Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 Restore EOC-RPT trip capability. OR B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours 2 hours
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump from service. OR C.2 Reduce THERMAL POWER to < ~~28%~~ RTP. 27.6%	4 hours 4 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

When a channel is placed ~~in an~~ inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days on an ALTERNATE TEST BASIS
SR 3.3.4.1.2	Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ ~~28%~~ RTP. 27.6%	24 months

(continued)

3.7 PLANT SYSTEMS

3.7.7 Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

OR

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.7.2	Perform a system functional test.	24 months
SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions[2] specified or incorporated below:

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not to exceed ~~2,768~~ megawatts thermal, in accordance with the conditions specified herein. 2,804

(2) <u>Technical Specifications</u>

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 168, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) <u>Additional Conditions</u>

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) <u>Fire Protection</u>

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained

2. The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2763 MWt. 2804
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ ~~25%~~ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.08 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

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

2.1.2 Reactor Coolant System (RCS) Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager, the corporate executive responsible for overall plant nuclear safety, and the offsite review committee.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ RTP. 24%

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ RTP. 24%	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~25%~~ RTP 24% <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~25%~~ 24% RTP AND 24 hours thereafter

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 28% RTP. 27.6%	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations.	12 hours
	AND		
	I.2	Restore required channels to OPERABLE.	120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1	Be in MODE 2.	4 hours

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, ~~entry into~~ associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	NOTE Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. 24% Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP. 24%	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ 28% RTP. ≥ 27.6%	24 months
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	---NOTES--- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.16	---NOTES--- 1. Neutron detectors are excluded. 2. (Not used.) 3. For Function 5, "*n*" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitor					
a. Neutron Flux - High	2	2(d)	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	2(d)	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	2(d)	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	2(d)	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitor					
a. Neutron Flux - High (Setdown)	2	3(c)	G	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Simulated Thermal Power - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	0.57W + 56.8% ~~≤ 0.58 W + 58%~~ RTP and ≤ 115.5% RTP(b)
c. Neutron Flux - High	1	3(c)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1, 2	3(c)	G	SR 3.3.1.1.10	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 58% − 0.58~~ ΔW RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) Each APRM channel provides inputs to both trip systems.

(d) One channel in each quadrant of the core must be OPERABLE whenever the IRMs are required to be OPERABLE. Both the RWM and a second licensed operator must verify compliance with the withdrawal sequence when less than three channels in any trip system are OPERABLE.

0.57W + 56.8% − 0.57

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Turbine Stop Valve - Closure	≥ ~~20%~~ RTP 27.6%	4	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≤ 10% closed
9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ ~~20%~~ RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16	≥ 600 psig
10. Reactor Mode Switch - Shutdown Position	1, 2	2	G	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
11. Manual Scram	1, 2	2	G	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine Trip High Water Level Instrumentation

LCO 3.3.2.2 Three channels of feedwater and main turbine trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ RTP.

24%

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater and main turbine high water level trip channels inoperable.	B.1 Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to < ~~25%~~ RTP.	4 hours

24%

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) - Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure - Low.

OR

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~28%~~ RTP.

≥ 27.6%

ACTIONS

---NOTE---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
	OR	
	A.2 ---NOTE--- Not applicable if inoperable channel is the result of an inoperable breaker. --- Place channel in trip.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1 Restore EOC-RPT trip capability.	2 hours
	OR	
	B.2 Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Remove the associated recirculation pump from service.	4 hours
	OR	
	C.2 Reduce THERMAL POWER to < ~~28%~~ RTP. 27.6%	4 hours

SURVEILLANCE REQUIREMENTS

--NOTE--
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days on an ALTERNATE TEST BASIS
SR 3.3.4.1.2	Verify TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is ≥ ~~28%~~ RTP. 27.6%	24 months

(continued)

3.7 PLANT SYSTEMS

3.7.7 Main Turbine Bypass System

LCO 3.7.7 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ RTP. ≥ 24%

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ RTP. [24%	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.7.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.7.2	Perform a system functional test.	24 months
SR 3.7.7.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints [LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"], in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR SL.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.
>
> 24%

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 5, 6, and 7). Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level.

24%

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

< 24%

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to in a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is

24%

reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

24%

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

24%

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

24%

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. FSAR, Chapter 3.
3. FSAR, Chapter 6.
4. FSAR, Chapter 14.
5. NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," August 1989.
6. NEDO-24395, "Load Line Limit Analysis," October 1980.
7. NEDC-30474-P "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E.I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.
11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level. 24%

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

24%

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 24% 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a 24% limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 24% 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.
24%

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems. 24%

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

24%

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

24%

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions specified in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals. The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses ≥ 28% RTP) 27.6% and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High (Setdown) (continued)

abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP. 24%

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP. 24%

The Average Power Range Monitor Neutron Flux - High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power - High

The Average Power Range Monitor Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux - High Function Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL 27.6% POWER ≥ ~~28%~~ RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if the TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~28%~~ RTP. This Function is not required when THERMAL POWER is < ~~28%~~ 27.6% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

27.6%

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 28% RTP. This Function is not required when THERMAL POWER is < 28% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

27.6%

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, directly to the scram pilot solenoid power circuits. These manual scram logic channels are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with two channels, each of which provides input into one of the RPS manual scram logic channels.

(continued)

BASES

SURVEILLANCE REUIREMENTS

SR 3.3.1.1.1 (continued)

between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

24%

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

24%

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

27.6%

POWER is ≥ 28% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 28% RTP to ensure that the calibration is valid.

27.6%

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 28% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 18.

SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that 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. For MSIV - Closure, SDV Water Level - High (Float Switch), and TSV - Closure Functions, this SR also includes a physical inspection and actuation of the switches. For the APRM Simulated Thermal Power - High Function, this SR also includes calibrating the associated recirculation loop flow channel.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the IRM SRs

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level setpoint, causing the trip of the two feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level - High instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high level trip indirectly initiates a reactor scram from the main turbine trip (above ~~28%~~ 27.6% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The LCO requires three channels of the Reactor Vessel Water Level - High instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.2. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions (nominal trip setpoint). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

24%

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ ~~25%~~ RTP to ensure that the specified acceptable fuel design limits are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below ~~25%~~ RTP; therefore, these requirements are only necessary when operating at or above this power level.

24%

(continued)

BASES

ACTIONS

B.1 (continued)

not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

24%

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < ~~25%~~ RTP within 4 hours. As discussed in the Applicability section of the Bases,
24% operation below ~~25%~~ RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < ~~25%~~ RTP from full power conditions in an orderly manner and without challenging plant systems.

24%

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2 and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 28% RTP.

27.6%

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

Turbine Stop Valve - Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink and increases reactor pressure, neutron flux, and heat flux that must be limited. Therefore, an RPT is initiated on a TSV - Closure signal before the TSVs are completely closed in anticipation of the effects that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each valve. While there are two separate position switches associated with each stop valve, only the signal from one switch for each TSV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

27.6%

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

27.6%

Turbine Stop Valve - Closure (continued)

27.6%

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 28% RTP. Below 28% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

27.6%

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 28% RTP. Below 28% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Neutron Flux - High Functions of the RPS are adequate to maintain the necessary margin to the MCPR SL.

27.6%

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

27.6%

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 28% RTP within 4 hours. Alternately, the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 28% RTP from full power conditions in an orderly manner and without challenging plant systems.

27.6%

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day on an ALTERNATE TEST BASIS Frequency is based on a review of the surveillance test history and Reference 8.

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure 27.6%
and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 28% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed 27.6%
during the calibration at THERMAL POWER ≥ 28% RTP to ensure that the calibration is valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 28% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met with the channel considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 7.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. 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. For the TSV - Closure Function, this SR also includes a physical inspection and actuation of the switches.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hour if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hour) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < ~~25%~~ 24% RTP.)

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can ~~occur~~ (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 10 of 11 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure ~~well~~ below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). ~~Sensitivity analyses have demonstrated that 8 or 9 S/RVs operating in the pressure relief mode will maintain the reactor vessel below 1375 psig.~~ This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event..

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND

During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases.

The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

APPLICABLE SAFETY ANALYSES

The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Section 9.4 and Appendix E (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2).

The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).

LCO

To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2763 MWt.
2804

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits [LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"] may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

24%

The Main Turbine Bypass System is required to be OPERABLE at ≥ ~~25%~~ RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure to maximum flow demand transient. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < ~~25%~~ RTP. Therefore, these requirements are only necessary when operating at or above this power level. 24%

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to

(continued)

BASES

ACTIONS

B.1 (continued)

24%

< ~~25%~~ RTP. As discussed in the Applicability section, operation at < ~~25%~~ RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection transient. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on a review of the surveillance test history and Reference 5.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in Technical Requirements Manual (Ref. 3). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

BASES

BACKGROUND (continued)

to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints [LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"], in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR Safety Limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28×10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28×10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

24%

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that certain fuel design limits identified in Reference 1 are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs), anticipated operational transients, and normal operation that determine the APLHGR limits are presented in References 1, 2, 3, 4, 5, 6, and 7.

Fuel design evaluations are performed to demonstrate that the 1% limit on the fuel cladding plastic strain and certain other fuel design limits described in Reference 1 are not exceeded during AOOs for operation with LHGRs up to the operating limit LHGR. APLHGR limits are equivalent to the LHGR limit for each fuel rod divided by the local peaking factor of the fuel assembly. APLHGR limits are developed as a function of exposure and the various operating core flow and power states to ensure adherence to fuel design limits during the limiting AOOs (Refs. 5, 6, and 7). Flow dependent APLHGR limits are determined (Ref. 7) using the three dimensional BWR simulator code (Ref. 8) to analyze slow flow runout transients. The flow dependent multiplier, $MAPFAC_f$, is dependent on the maximum core flow runout capability. The maximum runout flow is dependent on the existing setting of the core flow limiter in the Recirculation Flow Control System.

Based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions, power dependent multipliers, $MAPFAC_p$, are also generated. Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, both high and low core flow $MAPFAC_p$ limits are provided for operation at power levels between 25% RTP and the previously mentioned bypass power level.

24%

(continued)

BASES (continued)

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 7) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 25% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

24%

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to in a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to ≤ 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to ≤ 25% RTP in an orderly manner and without challenging plant systems.

24%

24%

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

24%

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

24%

REFERENCES

1. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel," (revision specified in the COLR).
2. FSAR, Chapter 4.
3. FSAR, Chapter 6.
4. FSAR, Chapter 15.
5. NEDO-24205, "E.I. Hatch Nuclear Plant Units 1 and 2 Single-Loop Operation," August 1989.
6. NEDO-24395, "Load Line Limit Analysis," October 1980.
7. NEDC-30474-P "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program for E.I. Hatch Nuclear Plant, Units 1 and 2," December 1983.
8. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
9. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.
10. NEDO-31376, "E.I. Hatch Nuclear Plant SAFER/GESTAR-LOCA Analysis," December 1986.
11. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined mainly by the one dimensional transient code (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

24%

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 25% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

24% 24% 24%

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on 24% operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems. 24%

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

24%

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. 24%

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

the specified Allowable Value, where appropriate. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Each channel must also respond within its assumed response time, where appropriate.

Allowable Values are specified for each RPS Function specified in the Table. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors.

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50 49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions specified in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals. The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses ≥ ~~28%~~ RTP) 27.6% and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High (Setdown) (continued)

abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed ~~25%~~ RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. 24% Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < ~~25%~~ RTP. 24%

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < ~~25%~~ RTP. 24%

The Average Power Range Monitor Neutron Flux - High (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High ~~Function~~ provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Simulated Thermal Power - High

The Average Power Range Monitor Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor Neutron Flux - High Function Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function.

27.6%

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if the TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~28%~~ RTP. This Function is not required when THERMAL POWER is < ~~28%~~ RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

27.6%

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (continued)

absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. 27.6%

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 28% RTP. This Function is not required when THERMAL POWER is < 28% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

27.6%

10. Reactor Mode Switch - Shutdown Position

The Reactor Mode Switch - Shutdown Position Function provides signals, via the manual scram logic channels, to each of the four RPS logic channels, which are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which provides input into one of the RPS logic channels.

(continued)

BASES

SURVILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

between instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

24%

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

24%

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 (continued)

POWER is ≥ ~~28%~~ 27.6% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ ~~28%~~ 27.6% RTP to ensure that the calibration is valid.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~28%~~ RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 20.

SR 3.3.1.1.13

~~A~~ CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies that 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. For MSIV - Closure, SDV Water Level - High (Float Switch), and TSV - Closure Functions, this SR also includes a physical inspection and actuation of the switches. For the APRM Simulated Thermal Power - High Function, this SR also includes calibrating the associated recirculation loop flow channel.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 effective full power hours LPRM calibration against the TIPs (SR 3.3.1.1.8). A second Note is provided that requires the IRM SRs

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The feedwater and main turbine high water level trip instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level setpoint, causing the trip of the two feedwater pump turbines and the main turbine.

Reactor Vessel Water Level - High signals are provided by level sensors that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level - High instrumentation are provided as input to a two-out-of-three initiation logic that trips the two feedwater pump turbines and the main turbine. The channels include electronic equipment (e.g., trip relays) that compare measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a main feedwater and turbine trip signal to the trip logic.

A trip of the feedwater pump turbines limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The high level trip indirectly initiates a reactor scram from the main turbine trip (above ~~28%~~ RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR. 27.6%

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

(continued)

BASES (continued)

LCO

The LCO requires three channels of the Reactor Vessel Water Level - High instrumentation to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Two of the three channels are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.2. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions (nominal trip setpoint). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

24%

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ ~~25%~~ RTP to ensure that the specified acceptable fuel design limits are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below ~~25%~~ RTP; therefore, these requirements are only necessary when operating at or above this power level.

24%

(continued)

BASES

ACTIONS

B.1 (continued)

not maintained). Therefore, continued operation is only permitted for a 2 hour period, during which feedwater and main turbine high water level trip capability must be restored. The trip capability is considered maintained when sufficient channels are OPERABLE or in trip such that the feedwater and main turbine high water level trip logic will generate a trip signal on a valid signal. This requires two channels to each be OPERABLE or in trip. If the required channels cannot be restored to OPERABLE status or placed in trip, Condition C must be entered and its Required Action taken.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to ≤ ~~25%~~ 24% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below ~~25%~~ 24% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to ≤ ~~25%~~ 24% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 2) assumption of the average time required to perform channel Surveillance. That analysis

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2 and 3.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit. The EOC-RPT function is automatically disabled when turbine first stage pressure is < ~~28%~~ RTP.

27.6%

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The setpoint is calibrated consistent with applicable setpoint methodology assumptions (nominal trip setpoint). Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

Allowable Values are specified for each EOC-RPT Function specified in the LCO. Nominal trip setpoints are specified in the setpoint calculations. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The nomial setspoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis in order to account for instrument uncertainties appropriate to the Function. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., TSV position), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

relay) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analysis, LCO, and Applicability discussions are listed below on a Function by Function basis.

Alternatively, since this instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

Turbine Stop Valve - Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink and increases reactor pressure, neutron flux, and heat flux that must be limited. Therefore, an RPT is initiated on a TSV - Closure signal before the TSVs are completely closed in anticipation of the effects that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each valve. While there are two separate position switches associated with each stop valve, only the signal from one switch for each TSV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function 27.6% must be enabled at THERMAL POWER ≥ 20% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

27.6%

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 28% RTP. Below 28% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR Safety Limit.

27.6%

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 28% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

27.6%

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 28% RTP. Below 28% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Neutron Flux - High Functions of the RPS are adequate to maintain the necessary margin to the MCPR SL.

27.6%

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

27.6%

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to ≤ ~~28%~~ RTP within 4 hours. Alternately, the associated recirculation pump may be removed from service, since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to ≤ ~~28%~~ RTP from full power conditions in an orderly manner and without challenging plant systems.

27.6%

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the recirculation pumps will trip when necessary.

(continued)

BASES

SUREVILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day on an ALTERNATE TEST BASIS Frequency is based on a review of the surveillance test history and Reference 8.

SR 3.3.4.1.2

27.6%

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 28% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed during the calibration at THERMAL POWER ≥ 28% RTP to ensure that the calibration is valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 28% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met with the channel considered OPERABLE.

27.6%

The 24 month Frequency is based on a review of the surveillance test history, drift of the associated instrumentation, and Reference 7.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. 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. For the TSV - Closure Function, this SR also includes a physical inspection and actuation of the switches.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Main Steam Line Pressure - Low

Low MSL pressure with the reactor at power indicates that there may be a problem with the turbine pressure regulation, which could result in a low reactor vessel water level condition and the RPV cooling down more than 100°F/hr if the pressure loss is allowed to continue. The Main Steam Line Pressure - Low Function is directly assumed in the analysis of the pressure regulator failure (Ref. 2). For this event, the closure of the MSIVs ensures that the RPV temperature change limit (100°F/hr) is not reached. In addition, this Function supports actions to ensure that Safety Limit 2.1.1.1 is not exceeded. (This Function closes the MSIVs prior to pressure decreasing below 785 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.)

24%

The MSL low pressure signals are initiated from four switches that are connected to the MSL header. The switches are arranged such that, even though physically separated from each other, each switch is able to detect low MSL pressure. Four channels of Main Steam Line Pressure - Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure - Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 2).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow - High

Main Steam Line Flow - High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow - High Function is directly assumed in the analysis of the main steam line break (MSLB) (Ref. 2). The isolation action, along with the scram function of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 100 limits.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of S/RVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS - Operating."

APPLICABLE SAFETY ANALYSIS

The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs), followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 1). For the purpose of the analyses, 10 of 11 S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure ~~well~~ below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). ~~Sensitivity analyses have demonstrated that 8 or 9 S/RVs operating in the pressure relief mode will maintain the reactor vessel below 1375 psig.~~ This LCO helps to ensure that the acceptance limit of 1375 psig is met during the Design Basis Event.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 Main Condenser Offgas

BASES

BACKGROUND	During unit operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser, then exhausted through the steam jet air ejectors (SJAEs) to the Main Condenser Offgas System. The offgas from the main condenser normally includes radioactive gases. The Main Condenser Offgas System has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.
APPLICABLE SAFETY ANALYSES	The main condenser offgas gross gamma activity rate is an initial condition of the Main Condenser Offgas System failure event, discussed in the FSAR, Sections 11.3 and 15.1.35 (Ref. 1). The analysis assumes a gross failure in the Main Condenser Offgas System that results in the rupture of the Main Condenser Offgas System pressure boundary. The gross gamma activity rate is controlled to ensure that, during the event, the calculated offsite doses will be well within the limits of 10 CFR 100 (Ref. 2). The main condenser offgas limits satisfy Criterion 2 of the NRC Policy Statement (Ref. 3).
LCO	To ensure compliance with the assumptions of the Main Condenser Offgas System failure event (Ref. 1), the fission product release rate should be consistent with a noble gas release to the reactor coolant of 100 μCi/MWt-second after decay of 30 minutes. This LCO is established consistent with this requirement (2436 MWt x 100 μCi/MWt-second = 240 mCi/second). The 240 mCi/second limit is conservative for a rated core thermal power of 2763 MWt. 2804

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits [LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"] may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ ~~25%~~ 24% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure to maximum flow demand transient. As discussed in the Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.2, sufficient margin to these limits exists at < ~~25%~~ 24% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to

(continued)

BASES

ACTIONS

B.1 (continued) 24%

< ~~25%~~ RTP. As discussed in the Applicability section, operation at < ~~25%~~ RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the turbine generator load rejection transient. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.7.1

Cycling each main turbine bypass valve through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.7.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 24 month Frequency is based on a review of the surveillance test history and Reference 5.

SR 3.7.7.3

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in Technical Requirements Manual (Ref. 3). The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

(continued)

Enclosure 5

Edwin I. Hatch Nuclear Plant
Request for Licensing Amendment
Measurement Uncertainty Recapture Power Uprate

Licensing Commitments

The following discussion specifies the Plant Hatch actions to which Southern Nuclear Operating Company (SNC) is committed as the result of the proposed measurement uncertainty recapture power uprate. Any other actions discussed within this submittal represent intended or planned actions by SNC and are provided for information only and are not considered regulatory commitments.

Plant Hatch Commitments

A. *Plant Modifications*

1. Implementation of the Crossflow system will be completed prior to implementation of the requested license amendment and prior to raising the rated thermal power (RTP) above 2763 MWt.

2. Validation of the assumed Crossflow system measurement uncertainty will be performed prior to implementation of the requested licensed amendment and prior to raising the power level above 2763 MWt.

B. *Administrative Changes*

1. Necessary maintenance and operational procedure revisions will be completed prior to implementation of the requested licensed amendment.

2. Operational procedure revisions will include the Crossflow system out-of-service administrative technical requirements.

C. *Startup Testing*

1. Core power from the average power range monitors (APRMs) will be rescaled to the uprated power level prior to exceeding the current licensed power level. Any necessary adjustments of the APRM alarm and trip settings will be made.

2. Demonstration of an acceptable fuel thermal margin will be performed prior to and during power ascension at each steady-state heat balance power level (95% and 100% of the current licensed power level and 100% of the uprated power level).

Fuel thermal margin will be projected to the uprated RTP point after the measurements at 95% and 100% of current licensed power level are taken to show the estimated margin. The demonstration of core and fuel conditions will be performed using current Plant Hatch methods.

3. In preparation for operation at the uprated power level, routine measurements of reactor and system pressures, flows, and selected major rotating equipment vibration will be taken near 95% and 100% of the current licensed power level and at 100% of the uprated power level.

4. The operational aspect of the uprate will be demonstrated by performing turbine pressure regulator controller and feedwater controller testing during power ascension testing. Reactor pressure control system testing, consistent with the guidelines of NEDC-33085P," Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," dated December 2002, will be performed during power ascension testing.

   During these test, a water level change of ± 3 inches and pressure setpoint changes of ± 3 psi will be used. If necessary, the controllers and actuator elements will be adjusted.

   a. The performance of the feedwater level control system will be recorded at 95% and 100% of the current licensed power level and confirmed at the uprated power lever during power ascension.

   b. The turbine pressure controller setpoint will be readjusted at 95% current licensed power level and held constant. Adjusting the pressure setpoint prior to recording the baseline power ascension data establishes a consistent basis for measuring the performance of the reactor and the turbine control valves.

D. *Training*

1. Minor changes (e.g., power/flow map and flow-reference setpoint changes) will be communicated through normal operator training.

2. Simulator changes and validation for the power uprate will be performed in accordance with ANSI/ANS 3.5-1985.

Enclosure 6

Edwin I. Hatch Nuclear Plant
Request for License Amendment
Measurement Uncertainty Recapture Power Uprate

NRC Regulatory Issue Summary 2002-03 Reconciliation

On January 31, 2002, the Nuclear Regulatory Commission issued "NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," which provides guidance on the content of measurement uncertainty recapture power uprate projects. Specifically, Attachment 1 of Regulatory Issue Summary 2002-03 provides a detailed breakdown of the specific subject matter that each licensee should address when applying for this type of power uprate.

To ensure all technical issues were appropriately addressed, Southern Nuclear Operating Company (SNC) performed the evaluations based upon the guidance of NEDC-32938P, "Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," dated July 2000. The results of all the evaluations performed are fully documented in NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," dated November 2002 (Enclosure 8).

Items II and III of Attachment 1 of NRC Regulatory Issue Summary 2002-03, identify the need for matrices identifying the bounded and nonbounded accident and transient analyses. Although Enclosure 8 does not specifically list matrices for these items, it presents the analyses performed or confirmed for each subject area. Within the section for each subject area, the existing analysis is identified as bounding or nonbounding. Where generic analyses from NEDC-32938P are used, the generic analysis is verified as being applicable to Plant Hatch within Enclosure 8. Where necessary, the specific sections present the results of the new analysis. All necessary accidents and transients applicable to Plant Hatch, which encompass the transients and accidents listed under these items in NRC Regulatory Issue Summary 2002-03, are evaluated in Enclosure 8.

Table E6-1 is a cross-reference of the guidance provided in NRC RIS 2002-03 to the documents (i.e., Enclosures 1, 2, 3, 4, 7, and 8 of this submittal) that provide the discussions of and/or justifications for the proposed Technical Specifications changes addressed herein.

TABLE E6-1

REQUIREMENTS ISSUE SUMMARY / SUBMITTAL CROSS REFERENCE

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty				
I.1	Detailed description of plant-specific implementation of feedwater flow measurement technique and power increase gained as result of implementing technique	E7	NA	Plant Modifications
II. Accidents and Transients for Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level				
II.1	Matrix for bounded accidents and transients	E8	All	NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," November 2002
III. Accidents and Transients for Which the Existing Analyses of Record Do Not Bound Plant Operation at the Proposed Uprated Power Level				
III.1, 2.A-D	Matrix for unbounded accidents and transients	E8	All	NEDC-33085P, "Safety Analysis Report for Edwin I. Hatch Units 1 and 2 THERMAL POWER OPTIMIZATION," November 2002

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV. Mechanical/Structural/Material Component Integrity and Design				
IV.1.A.i	Reactor vessel, nozzles, supports	E8	3.2	3.2 Reactor Vessel 3.2.1 Fracture Toughness 3.2.2 Reactor Vessel Structural Evaluation 3.2.2.1 Design Conditions 3.2.2.2 Normal and Upset Conditions 3.2.2.3 Emergency and Faulted Conditions
IV.1.A.ii	Reactor core support structures and vessel internals	E8	3.3 and 3.4	3.3. Reactor Internals 3.3.1. Reactor Internal Pressure Difference 3.3.2. Reactor Internal Structural Evaluation 3.3.3. Steam Separator and Dryer Performance 3.4 Flow-Induced Vibration
IV.1.A.iii	Control rod drive mechanisms	E8	2.5	2.5. Reactivity Control

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	E8	3.4, 3.5, 3.6, 3.7, 3.8, 3.9, 3.10, and 3.11	3.4 Flow-Induced Vibration 3.5 Piping Evaluation 3.5.1 Reactor Coolant Press Boundary Piping 3.6 Reactor Recirculation System 3.7 Main Steam Line Flow Restrictors 3.8 Main Steam Isolation Valves 3.9 Reactor Core Isolation Cooling 3.10 Residual Heat Removal System 3.11 Reactor Water Cleanup System
IV.1.A.v	Balance of plant (BOP) piping (NSSS interface systems, safety related cooling water systems, and containment systems)	E8	3.5	3.5. Piping Evaluation 3.5.2. Balance-of-Plant Piping Evaluation
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	NA	NA	NA
IV.1.A.vii	Reactor coolant pumps	NA	NA	NA
IV.1.A.viii	Pressurizer shell, nozzles, and surge lines	NA	NA	NA

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV.1.A.ix	Safety-related valves	E8	3.1, 3.8, 4.1, and 6.5	3.1. Nuclear System Pressure Relief/Over Pressure Protection 3.8. Main Steam Isolation Valves 4.1. Containment System Performance 4.1.2. Generic Letter 95-07 Program 6.5. Standby Liquid Control System
IV.1.B.i	Stresses	E8	3.2, 3.4, 3.5, and 7.1	3.2. Reactor Vessel 3.2.2. Reactor Vessel Structural Evaluation 3.4. Flow-Induced Vibration 3.5. Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2 Balance of Plant Piping Evaluation 7.1 Turbine Generator
IV.1.B.ii	Cumulative usage factors	E8	3.2	3.2. Reactor Vessel 3.2.2 Reactor Vessel Structural Evaluation
IV.1.B.iii	Flow induced vibration	E8	3.4	3.4. Flow-Induced Vibration

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV.1.B.iv	Changes in temperature (pre- and post-uprate)	E8	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
IV.1.B.v	Changes in pressure (pre-and post-uprate)	E8	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
IV.1.B.vi	Changes in flow rate (pre- and post-uprate)	E8	1.3	1.3 TPO Plant Operating Conditions 1.3.1 Reactor Heat Balance 1.3.2 Reactor Performance Improvement Features
IV.1.B.vii	High-energy line break locations	E8	10.1	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV.1.B.viii	Jet impingement and thrust forces	E8	10.1, and 10.2	10.1 High Energy Line Break 10.1.1 Steam Line Breaks 10.1.2 Liquid Line Breaks 10.1.2.6 Pipe Whip and Jet Impingement 10.2 Moderate Energy Line Break
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	E8	3.1	3.1 Nuclear System Pressure Relief/Overpressure Protection
IV.1.C.ii	Reactor vessel fluence evaluation	E8	3.2	3.2. Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure-temperature limit curves	E8	3.2	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.iv	Reactor vessel low-temperature overpressure protection	NA	NA	NA
IV.1.C.v	Reactor vessel upper shelf energy	E8	3.2	3.2 Reactor Vessel 3.2.1 Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	E8	3.2	3.2 Reactor Vessel 3.2.1 Fracture Toughness

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
IV.1.D	Code of record	E8	3.2, and 3.5	3.2.2 Reactor Vessel Structural Evaluation 3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping 3.5.2. Balance-of-Plant Piping Evaluation
IV.1.E	Component inspection/testing programs and erosion/corrosion programs	E8	3.5 and 10.6	3.5 Piping Evaluation 3.5.1 Reactor Coolant Pressure Boundary Piping Evaluation (including flow-accelerated corrosion and piping inspection programs) 3.5.2 Balance-of-Plant Piping Evaluation (including flow-accelerated corrosion and piping inspection programs) 10.6 Plant Life
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	NA	NA	NA
V. Electrical Equipment Design				
V.1.A	Emergency diesel generators	E8	6.1	6.1 AC Power 6.1.2 On-Site Power

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
V.1.B	Station blackout equipment	E8	9.3	9.3.2 Station Blackout
V.1.C	Environmental qualification of electrical equipment	E8	10.3	10.3 Environmental Qualification 10.3.1 Electrical Equipment
V.1.D	Grid stability	E8	6.1	6.1 AC Power 6.1.1 Off-Site Power

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VI. System Design				
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling, as applicable)	E8	3.4 through3.11	3.4 Flow-Induced Vibration 3.5 Piping Evaluation 3.5.1 Reactor Coolant Boundary Piping 3.5.2 Balance-of-Plant Piping 3.6 Reactor Recirculation System 3.7 Main Steam Line Flow Restrictors 3.8 Main Steam Isolation Valves 3.9 Reactor Core Isolation Cooling 3.10 Residual Heat Removal 3.11 Reactor Water Cleanup
VI.1.B	Containment systems	E8	4.1	4.1 Containment System Performance 4.1.1 Generic Letter 89-10 Program 4.1.2 Generic Letter 95-07 Program 4.1.3 Generic Letter 96-06

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VI.1.C	Safety-related cooling water systems	E8	6.4	6.4. Water Systems 6.4.1. Service Water Systems 6.4.5. Ultimate Heat Sink
VI.1.D	Spent fuel pool storage and cooling systems	E8	6.3	6.3 Fuel Pool 6.3.1 Fuel Pool Cooling 6.3.2 Crud Activity and Corrosion Products 6.3.3 Radiation Levels 6.3.4 Fuel Racks
VI.1.E	Radioactive waste systems	E8	4.5, and 8.1 through 8.6	4.5 Standby Gas Treatment System 8.1 Liquid and Solid Waste Management 8.2 Gaseous Waste Management 8.3 Radiation Sources in the Reactor Core 8.4 Radiation Sources in the Reactor Coolant 8.4.1 Coolant Activation Products 8.4.2 Activated Corrosion and Fission Products 8.5 Radiation Levels 8.6 Normal Operation Off-Site Doses

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VI.1.F	Engineered safety features (ESF) heating, ventilation, and air conditioning systems	E8	4.4, 4.6, 6.4, and 6.6	4.4 Main Control Room Atmosphere Control System 4.6 Post-LOCA Combustible Gas Control 6.4.4 Chilled Water Systems 6.6 Power Dependent Heating, Ventilation, and Air Conditioning
VII. Other				
VII.1	Operator actions and effects on time available	E8	4.1, 6.7, 9.3, 10.5, and 10.8	4.1 Containment System Performance 6.7 Fire Protection 9.3 Special Events 9.3.2 Station Blackout 10.5 Operator Training and Human Factors 10.8 Emergency Operating Procedures
VII.2.A	Emergency and abnormal operating procedures	E7	NA	Plant Modifications
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	E7	NA	Plant Modifications

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VII.2.C	The control room plant reference simulator	E7	NA	Plant Modifications
VII.2.D	The operator training program	E7	NA	Plant Modifications
VII.3	Licensee intent to complete modifications identified in item 2 above (including training of operators), prior to implementation of the power uprate	E7	NA	Plant Modifications
VII.4	Licensee intent to revise plant operating procedures related to temporary operation above “full steady-state licensed power levels” to reduce the magnitude of the allowed deviation from the licensed power level (reduce from 2% to 0.5%)	E7	NA	Plant Modifications
VII.5.A	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on types and amounts of effluents released offsite, and whether bounded by final environmental statement and previous Environmental Assessments for the plant	E1	NA	Environmental Assessment

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VII.5.B	10 CFR51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on individual and cumulative occupational radiation exposure	E1	NA	Environmental Assessment
VIII. Changes to Technical Specifications, Protection System Settings, Emergency System Settings				
VIII.1	A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate	E2	NA	Request to Revise Operating Licenses and Technical Specifications; and Summary of Changes to Protection/Emergency System Settings
VIII.1.A	Description of the change	E2, E3, and E4	NA	E2 Request to Revise Operating Licenses and Technical Specifications; and Summary of Changes to Protection/Emergency System Settings E3 Page Change Instructions E4 Operating License, Technical Specifications, and Bases Markups

TABLE E6-1 (Continued)

NRC RIS Item No.	Description	Submittal Enclosure No.	Eclosure-8 NEDC-33085P Section	Enclosure Description/Section Details
VIII.1.B	Description of the analyses affected by and/or supporting the change	E2	NA	Request to Revise Operating Licenses and Technical Specifications; and Summary of Changes to Protection/Emergency System Settings
VIII.1.C	Justification for the change, including the type of information discussed in Section III above, for any analyses that support and/or are affected by change	E2	NA	Request to Revise Operating Licenses and Technical Specifications; and Summary of Changes to Protection/Emergency System Settings