Exelon Generation.

4300 Winfield Road
Warrenville, IL 60555
630 657 2000 Office

RS-19-003

10 CFR 50.90

February 1, 2019

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

James A. FitzPatrick Nuclear Power Plant, Unit 1
Renewed Facility Operating License No. DPR-59
Docket No. 50-333

LaSalle County Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Limerick Generating Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-39 and NPF-85
NRC Docket Nos. 50-352 and 50-353

Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
NRC Docket Nos. 50-410

Peach Bottom Atomic Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Application to Adopt TSTF-564, "Safety Limit MCPR"

References:
1. TSTF-564, Revision 2, "Safety Limit MCPR," dated October 24, 2018
2. Final Safety Evaluations of Technical Specifications Task Force Traveler TSTF-564, Revision 2 "Safety Limit MCPR," using the Consolidated Line Item Improvement Process, dated November 16, 2018 (CAC No. MG0161, EPID L-2017-PMP-0007)

In accordance with the provisions of 10 CFR 50.90 "Application for amendment of license, construction permit, or early site permit," Exelon Generation Company, LLC (EGC) is submitting a request for an amendment to:

Facility Operating License (FOL) No. NPF-62 for Clinton Power Station (CPS), Unit 1;
Renewed FOL No. DPR-59 for James A. FitzPatrick Nuclear Power Plant (JAF), Unit 1;
Renewed FOL Nos. NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2;
Renewed FOL Nos. NPF-39 and NPF-85 for Limerick Generating Station (LGS), Units 1 and 2;
Renewed FOL No. NPF-69 for Nine Mile Point Nuclear Station (NMP), Unit 2;
and Renewed FOL Nos. DPR-44 and DPR-56 for Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3.

EGC requests adoption of TSTF-564, "Safety Limit MCPR," Revision 2 (Reference 1), which is an approved change to the Improved Standard Technical Specifications (ISTS) (Reference 2), into each Unit's Technical Specifications. The proposed amendment revises the Technical Specifications (TS) safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for an SL. The proposed changes for LGS, Units 1 and 2, include additional revisions to MCPR references to also reduce the need for cycle-specific changes.

The proposed changes have been reviewed and recommended for approval by the affected Plant Operations Review Committees in accordance with the EGC Quality Assurance Program.

Attachment 1 provides a description and assessment of the proposed changes. Attachments 2a through 2f provide the existing TS pages marked to show the proposed changes. Attachments 3a through 3f provide revised (clean) TS pages. Attachments 4a through 4f provide existing TS Bases pages marked to show the proposed changes for information only.

Approval of the proposed amendment is requested by August 1, 2019, in order to support the Clinton Power Station, Unit 1 Refueling Outage that is planned for September 2019. Once approved, the amendments shall be implemented according to the following table, prior to entering Mode 4 following the specified Refueling Outage (Calendar Year (CY) provided for reference):

Unit(s)	Date (Prior to entering Mode 4)
CPS, Unit 1	Refueling Outage C1R19 (CY 2019)
JAF, Unit 1	Refueling Outage FPR24 (CY 2020)
LSCS, Units 1 and 2	Refueling Outage L1R18 (CY 2020)
LGS, Unit 1	Refueling Outage Li1R18 (CY 2020)
LGS, Unit 2	Refueling Outage Li2R16 (CY 2021)
NMP, Unit 2	Refueling Outage N2R17 (CY 2020)
PBAPS, Unit 2	Refueling Outage P2R23 (CY 2020)
PBAPS, Unit 3	Refueling Outage P3R22 (CY 2019)

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State Officials.

There are no regulatory commitments contained within this letter. Should you have any questions concerning this letter, please contact Mr. Ryan Sprengel at (630) 657-2814.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 1st day of February 2019.

Respectfully,

David M. Gullott
Director – Licensing
Exelon Generation Company, LLC

Attachments:
1. Description and Assessment
2a. Proposed Technical Specifications Changes (Mark-Up) for Clinton Power Station, Unit 1
2b. Proposed Technical Specifications Changes (Mark-Up) for James A. FitzPatrick Nuclear Power Plant, Unit 1
2c. Proposed Technical Specifications Changes (Mark-Up) for LaSalle County Station, Units 1 and 2
2d. Proposed Technical Specifications Changes (Mark-Up) for Limerick Generating Station, Units 1 and 2
2e. Proposed Technical Specifications Changes (Mark-Up) for Nine Mile Point Nuclear Station, Unit 2
2f. Proposed Technical Specifications Changes (Mark-Up) for Peach Bottom Atomic Power Station, Units 2 and 3
3a. Revised Technical Specifications Pages for Clinton Power Station, Unit 1
3b. Revised Technical Specifications Pages for James A. FitzPatrick Nuclear Power Plant, Unit 1
3c. Revised Technical Specifications Pages for LaSalle County Station, Units 1 and 2
3d. Revised Technical Specifications Pages for Limerick Generating Station, Units 1 and 2
3e. Revised Technical Specifications Pages for Nine Mile Point Nuclear Station, Unit 2
3f. Revised Technical Specifications Pages for Peach Bottom Atomic Power Station, Units 2 and 3
4a. Proposed Technical Specifications Bases Changes (Mark-Up) for Clinton Power Station, Unit 1 (For Information Only)
4b. Proposed Technical Specifications Bases Changes (Mark-Up) for James A. FitzPatrick Nuclear Power Plant, Unit 1 (For Information Only)

4c. Proposed Technical Specifications Bases Changes (Mark-Up) for LaSalle County Station, Units 1 and 2 (For Information Only)
4d. Proposed Technical Specifications Bases Changes (Mark-Up) for Limerick Generating Station, Units 1 and 2 (For Information Only)
4e. Proposed Technical Specifications Bases Changes (Mark-Up) for Nine Mile Point Nuclear Station, Unit 2 (For Information Only)
4f. Proposed Technical Specifications Bases Changes (Mark-Up) for Peach Bottom Atomic Power Station, Units 2 and 3 (For Information Only)

cc: NRC Regional Administrator, Region I
NRC Regional Administrator, Region III
NRC Senior Resident Inspector – Clinton Power Station
NRC Senior Resident Inspector – James A. FitzPatrick Nuclear Power Plant
NRC Senior Resident Inspector – LaSalle County Station
NRC Senior Resident Inspector – Limerick Generating Station
NRC Senior Resident Inspector – Nine Mile Point Nuclear Station
NRC Senior Resident Inspector – Peach Bottom Atomic Power Station
NRC Project Managers – NRR (CPS, JAF, LSCS, LGS, NMP, PBAPS, and Fleet)
Illinois Emergency Management Agency – Division of Nuclear Safety
Director, Bureau of Radiation Protection – Pennsylvania Department of Environmental Protection
A. L. Peterson, NYSERDA
R.R. Janati, Pennsylvania Bureau of Radiation Protection
D. Tancabel, State of Maryland

1.0 DESCRIPTION

Exelon Generation Company, LLC (EGC) requests adoption of TSTF-564, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Clinton Power Station (CPS), Unit 1; James A. FitzPatrick Nuclear Power Plant (JAF), Unit 1; LaSalle County Station (LSCS), Units 1 and 2; Limerick Generating Station (LGS), Units 1 and 2; Nine Mile Point Nuclear Station (NMP), Unit 2; and Peach Bottom Atomic Power Station (PBAPS), Units 2 and 3 Technical Specifications (TS). The proposed amendment revises the Technical Specification (TS) safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for an SL. The proposed changes for LGS, Units 1 and 2, include additional revisions to MCPR references to also reduce the need for cycle-specific changes.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

EGC has reviewed the safety evaluation for TSTF-564 provided to the Technical Specifications Task Force in a letter dated November 16, 2018. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-564. EGC has concluded that the justifications presented in TSTF-564 and the safety evaluation prepared by the NRC staff are applicable to CPS, Unit 1; JAF, Unit 1; LSCS, Units 1 and 2; LGS, Units 1 and 2; NMP, Unit 2; and PBAPS, Units 2 and 3 and justify this amendment for the incorporation of the changes to the CPS, JAF, LSCS, LGS, NMP, and PBAPS TS.

The CPS, Unit 1, reactor is currently fueled with GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The JAF, Unit 1, reactor is currently fueled with GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The LSCS, Units 1 and 2, reactors are currently fueled with GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The LGS, Units 1 and 2, reactors are currently fueled with GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The NMP, Unit 2, reactor is currently fueled with GE14 and GNF2 fuel bundles, for cores loaded with a mix of applicable fuel types the $SLMCPR_{95/95}$ is based on the largest $MCPR_{95/95}$ value for the fuel types used. The NMP, Unit 2, reactor is planned to be fueled with only GNF2 fuel bundles following the N2R17 refueling outage in 2020. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The PBAPS, Units 2 and 3, reactors are currently fueled with GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The MCPR value calculated as the point at which 99.9% of the fuel rods would not be susceptible to boiling transition (i.e., reduced heat transfer) during normal operation and anticipated operational occurrences is referred to as $MCPR_{99.9\%}$. Technical Specification 5.6.3, "Core Operating Limits Report (COLR)," is revised to require the $MCPR_{99.9\%}$ value to be included in the cycle-specific COLR.

2.2 Variations

EGC is proposing the following variations from the TS changes described in TSTF-564 or the applicable parts of the NRC staff's safety evaluation dated November 16, 2018.

The traveler and Safety Evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC).

2.2.1 Clinton Power Station

The CPS TS utilize different numbering than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 5.6.5 CORE OPERATING LIMITS REPORT (COLR) of the CPS TS is numbered 5.6.3 in TSTF-564. This difference is administrative and does not affect the applicability of TSTF-564 to the CPS TS.

The CPS TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.1.2 and Applicability in TS 3.2.2, but these differences do not affect the applicability of the TSTF-564 justification.

2.2.2 James A. FitzPatrick Nuclear Power Plant

The JAF TS utilize different numbering than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 5.6.5 CORE OPERATING LIMITS REPORT (COLR) of the JAF TS is numbered 5.6.3 in TSTF-564. This difference is administrative and does not affect the applicability of TSTF-564 to the JAF TS.

The JAF TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.1.2, but this difference does not affect the applicability of the TSTF-564 justification.

JAF was not licensed to the 10 CFR 50, Appendix A, GDC. The JAF equivalents of the referenced GDC are located in section 1.5 of the JAF Updated Final Safety Analysis Report (UFSAR). This difference does not alter the conclusion that the proposed change is applicable to JAF.

2.2.3 LaSalle County Station

The LSCS TS utilize different numbering than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 5.6.5 CORE OPERATING LIMITS REPORT (COLR) of the LSCS TS is numbered 5.6.3 in TSTF-564. This difference is administrative and does not affect the applicability of TSTF-564 to the LSCS TS.

The LSCS TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.1.2, but this difference does not affect the applicability of the TSTF-564 justification.

2.2.4 Limerick Generating Station

The LGS TS utilize different numbering and titles than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 2.1.2 THERMAL POWER, High Pressure and High Flow of the LGS TS is numbered 2.1.1.2 in TSTF-564. Section 3/4.2.3 MINIMUM CRITICAL POWER RATIO of the LGS TS is numbered 3.2.2 in TSTF-564.Section 6.9.1.9 CORE OPERATING LIMITS REPORT (COLR) of the LGS TS is numbered 5.6.3 in TSTF-564. These differences are administrative and do not affect the applicability of TSTF-564 to the LGS TS.

The LGS TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.2, but this difference does not affect the applicability of the TSTF-564 justification.

Three additional changes are included specific to LGS, Units 1 and 2. Specific values for operating limit MCPR in section 3/4.1.4.3 ROD BLOCK MONITOR are proposed to be changed to point to these limits specified in the COLR. A corresponding change was identified during this review for section 6.9.1.9 CORE OPERATING LIMITS REPORT, subsection f, clarifying the Technical Specification sections connected to the Rod Block Monitor setpoints and MCPR operability limits, respectively. A Technical Specifications Bases change in section 3/4.4 REACTOR COOLANT SYSTEM revises the current reference to the "MCPR fuel cladding safety limit" to "MCPR(99.9%)." These changes are administrative in nature.

2.2.5 Nine Mile Point Nuclear Station

The NMP Unit 2 TS utilize different numbering than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 5.6.5 CORE OPERATING LIMITS REPORT (COLR) of the NMP Unit 2 TS is numbered 5.6.3 in TSTF-564. This difference is administrative and does not affect the applicability of TSTF-564 to the NMP Unit 2 TS.

The NMP Unit 2 TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.1.2 and Applicability in TS 3.2.2, but these differences do not affect the applicability of the TSTF-564 justification.

2.2.6 Peach Bottom Atomic Power Station

The PBAPS TS utilize different numbering and titles than the Standard Technical Specifications on which TSTF-564 was based. Specifically, section 5.6.5 CORE OPERATING LIMITS REPORT (COLR) of the PBAPS TS is numbered 5.6.3 in TSTF-564. This difference is administrative and does not affect the applicability of TSTF-564 to the PBAPS TS.

The PBAPS TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.2 and Applicability in TS 3.2.2, but these differences do not affect the applicability of the TSTF-564 justification.

PBAPS was not licensed to the 10 CFR 50, Appendix A, GDC. The PBAPS UFSAR, Appendix H, "Conformance to AEC (NRC) Criteria," contains an evaluation of the design bases of PBAPS with respect to the GDC proposed to be added to 10 CFR 50 as Appendix A in July 1967. This difference does not alter the conclusion that the proposed change is applicable to PBAPS.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

EGC requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the CPS, Unit 1; JAF, Unit 1; LSCS, Units 1 and 2; LGS, Units 1 and 2; NMP, Unit 2; and PBAPS, Units 2 and 3 Technical Specifications (TS). The proposed change revises the Technical Specifications (TS) safety limit on minimum critical power ratio (SLMCPR). The revised limit calculation method is based on using the Critical Power Ratio (CPR) data statistics and is revised from ensuring that 99.9% of the rods would not be susceptible to boiling transition to ensuring that there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling. A single SLMCPR value will be used instead of two values applicable when one or two recirculation loops are in operation. TS 5.6.3, "Core Operating Limits Report (COLR)," is revised to require the current SLMCPR value to be included in the COLR. The request for LGS, Units 1 and 2, include a revision to 3/4.4 REACTOR COOLANT SYSTEM to point to MCPR limits specified in the COLR and to 6.9.1.9 to clarify the reference Specifications.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the Core Operating Limits Report (COLR). The SLMCPR is not an initiator of any accident previously evaluated. The revised safety limit values continue to ensure for all accidents previously evaluated that the fuel cladding will be protected from failure due to transition boiling. The proposed amendment for LGS, Units 1 and 2, also includes a revision to point to MCPR limits specified in the COLR and to clarify reference Specifications. The proposed change does not affect plant operation or any procedural or administrative controls on plant operation that affect the functions of preventing or mitigating any accidents previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?**

   Response: No

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. The proposed amendment for LGS, Units 1 and 2, also includes a revision to point to MCPR limits specified in the COLR and to clarify reference Specifications. The proposed change will not affect the design function or operation of any structures, systems or components (SSCs). No new equipment will be installed. As a result, the proposed change will not create any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Does the proposed amendment involve a significant reduction in a margin of safety?**

   Response: No

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. The proposed amendment for LGS, Units 1 and 2, also includes a revision to point to MCPR limits specified in the COLR and to clarify reference Specifications. This will result in a change to a safety limit, but will not result in a significant reduction in the margin of safety provided by the safety limit. As discussed in the application, changing the SLMCPR methodology to one based on a 95% probability with 95% confidence that no fuel rods experience transition boiling during an anticipated transient instead of the current limit based on ensuring that 99.9% of the fuel rods are not susceptible to boiling transition does not have a significant effect on plant response to any analyzed accident. The SLMCPR and the TS Limiting Condition for Operation (LCO) on MCPR continue to provide the same level of assurance as the current limits and do not reduce a margin of safety.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, EGC concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

ATTACHMENT 2a
Proposed Technical Specifications Changes (Mark-Up) for
Clinton Power Station, Unit 1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ ~~1.09~~1.07~~ for two recirculation loop operation or ≥ 1.12 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 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 and the corporate executive responsible for overall plant nuclear safety.

(continued)

2.0 SLs

2.2 SL Violations (continued)

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the plant manager and the corporate executive responsible for overall plant nuclear safety.

2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.

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 ≥ 21.6% 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 < 21.6% 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 ≥ 21.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),

2. LCO 3.2.2, Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$,

3. LCO 3.2.3, Linear Heat Generation Rate (LHGR),

4. LCO 3.3.1.1, RPS Instrumentation (SR 3.3.1.1.14),

5. LCO 3.3.1.3, Oscillation Power Range Monitor (OPRM) Instrumentation, and

6. LCO 3.7.6, Main Turbine Bypass System, (cycle dependent thermal power limits for an inoperable Main Turbine Bypass System).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in

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

(2) NEDO-32465, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications."

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

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

(continued)

ATTACHMENT 2b
Proposed Technical Specifications Changes (Mark-Up) for
James A. FitzPatrick Nuclear Power Plant, Unit 1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 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 Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

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% 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.	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 AND In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements (continued)

5.6.4 Not Used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) of Specification 3.2.1;
2. The MINIMUM CRITICAL POWER RATIO (MCPR), and MCPR(99.9%) of Specification 3.2.2;
3. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.3;
4. The Reactor Protection System (RPS) APRM Neutron Flux - High (Flow Biased) Function Allowable Value of Table 3.3.1.1·1;
5. The Rod Block Monitor - Upscale Function Allowable Value of Table 3.3.2.1- 1; and
6. The Power/Flow Exclusion Region of Specification 3.4.1.

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

1. NEDE- 24011-P·A, General Electric Standard Application for Reactor Fuel;
2. NEDC-3 1317P, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR- LOCA Loss-of-Coolant Accident Analysis: and
3. NED0-31960-A, BWR Owners ' Group Long-Term Stability Solutions Licensing Methodology.

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

No Changes. For Information Only.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

5.6.6 PAM Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
   i) Limiting Conditions for Operation Section 3.4.9 "RCS Pressure and Temperature (P/T) Limits"
   ii) Surveillance Requirements Section 3.4.9 "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
   i) SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors"
   ii) SIA Calculation 0800846.301, "2" Instrument Nozzle Stress Analysis"

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

ATTACHMENT 2c
Proposed Technical Specifications Changes (Mark-Up) for
LaSalle County Station, Units 1 and 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

~~For Unit 1,~~ MCPR shall be ≥ 1.07~~1.11 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation~~.

~~For Unit 2, MCPR shall be ≥ 1.12 for two recirculation loop operation or ≥ 1.15 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 Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

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% 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.	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 AND In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

-------------------------------NOTE-------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 (Deleted)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The APLHGR for Specification 3.2.1.

2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. The Rod Block Monitor Upscale Instrumentation Setpoint for the Rod Block Monitor—Upscale Function Allowable Value for Specification 3.3.2.1.

5. The OPRM setpoints for the trip function for SR 3.3.1.3.3.

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

1. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

2. ANF-913(P)(A), "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis."

3. ANF-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option."

4. XN-NF-80-19(P)(A), "Advanced Nuclear Fuel Methodology for Boiling Water Reactors."

5. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel."

6. EMF-CC-074(P)(A), Volume 4 - "BWR Stability Analysis: Assessment of STAIF with input from MICROBURN-B2."

7. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model."

8. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. EMF-2209(P)(A), "SPCB Critical Power Correlation."
10. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs."
11. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."
12. NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."
13. EMF-85-74(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."
14. EMF-2158(P)(A), "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2."
15. NEDC-33106P, "GEXL97 Correlation for Atrium-10 Fuel."
16. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel."
17. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model."
18. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996.
19. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

ATTACHMENT 2d
Proposed Technical Specifications Changes (Mark-Up) for
Limerick Generating Station, Units 1 and 2

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ~~1.10 for two recirculation loop operation and shall not be less than 1.14 for single recirculation loop operation~~1.07 with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than ~~1.10 for two recirculation loop operation or less than 1.14 for single recirculation loop operation~~1.07 and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ~~1.10 for two recirculation loop operation and shall not be less than 1.14 for single recirculation loop operation~~1.07 with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than ~~1.10 for two recirculation loop operation or less than 1.14 for single recirculation loop operation~~1.07 and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than ~~1.70~~the limit specified in the CORE OPERATING LIMITS REPORT (COLR), or THERMAL POWER greater than or equal to 90% of rated with MCPR less than ~~1.40~~the limit specified in the COLR.

ACTION:

a. With one RBM channel inoperable:

1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than ~~1.70~~the limit specified in the CORE OPERATING LIMITS REPORT (COLR), or THERMAL POWER greater than or equal to 90% of rated with MCPR less than ~~1.40~~the limit specified in the COLR.

ACTION:

a. With one RBM channel inoperable:

1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

****NO CHANGES THIS PAGE****
(INFORMATION ONLY)

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the rated MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2 and the main turbine bypass system is OPERABLE per Specification 3.7.8, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

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

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2a, and during reactor startups prior to control rod scram time tests in accordance with Specification 4.1.3.2.b.1.b, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit, including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

a. In accordance with the Surveillance Frequency Control Program,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

****NO CHANGES THIS PAGE****
(INFORMATION ONLY)

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the rated MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, provided that the end-of-cycle recirculation pump trip (EOC-RPT) system is OPERABLE per Specification 3.3.4.2 and the main turbine bypass system is OPERABLE per Specification 3.7.8, with:

$$\tau = \frac{(\tau_{ave} - \tau_B)}{\tau_A - \tau_B}$$

where:

τ_A = 0.86 seconds, control rod average scram insertion time limit to notch 39 per Specification 3.1.3.3,

$$\tau_B = 0.672 + 1.65 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{1/2} (0.016),$$

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i},$$

n = number of surveillance tests performed to date in cycle,

N_i = number of active control rods measured in the i^{th} surveillance test,

τ_i = average scram time to notch 39 of all rods measured in the i^{th} surveillance test, and

N_1 = total number of active rods measured in Specification 4.1.3.2.a.

APPLICABILITY:

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

****NO CHANGES THIS PAGE****
(INFORMATION ONLY)

LIMITING CONDITION FOR OPERATION (Continued)

ACTION

a. With the end-of-cycle recirculation pump trip system inoperable per Specification 3.3.4.2, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) EOC-RPT inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

b. With MCPR less than the applicable MCPR limit adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

c. With the main turbine bypass system inoperable per Specification 3.7.8, operation may continue provided that, within 1 hour, MCPR is determined to be greater than or equal to the rated MCPR limit as a function of the average scram time (shown in the CORE OPERATING LIMITS REPORT) main turbine bypass valve inoperable curve, adjusted by the MCPR(P) and MCPR(F) factors as shown in the CORE OPERATING LIMITS REPORT.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR, with:

a. τ = 1.0 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2a and during reactor startups prior to control rod scram time tests in accordance with Specification 4.1.3.2.b.1.b, or

b. τ as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit including application of the MCPR(P) and MCPR(F) factors as determined from the CORE OPERATING LIMITS REPORT.

a. In accordance with the Surveillance Frequency Control Program,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and in accordance with the Surveillance Frequency Control Program when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factors for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification ~~3.3.6~~3.1.4.3,

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

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

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),*

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

---

* For Cycle 8, specific documents were approved in the Safety Evaluation dated (5/4/98) to support License Amendment No. (127).

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factor for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification ~~3.3.6~~3.1.4.3.

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

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

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

ATTACHMENT 2e
Proposed Technical Specifications Changes (Mark-Up) for
Nine Mile Point Nuclear Station, Unit 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ ~~1.17 for two recirculation loop operation or ≥ 1.17 for single recirculation loop operation~~1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

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 ≥ 23% 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 < 23% 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 ≥ 23% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.2 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.4

5.6.2 Annual Radiological Environmental Operating Report (continued)

and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

-- NOTE --
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. The APLHGR for Specification 3.2.1.

2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

4. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power - High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

5. The Allowable Values, NTSPs, and MCPR conditions for the Rod Block Monitor – Upscale Functions for Specification 3.3.2.1.

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

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, (NRC approved version specified in the COLR).

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

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

(continued)

ATTACHMENT 2f
Proposed Technical Specifications Changes (Mark-Up) for
Peach Bottom Atomic Power Station, Units 2 and 3

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07~~1.15 for two recirculation loop operation or ≥ 1.15 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 Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

~~(continued)~~

~~2.0 SLs~~

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 ≥ 22.6% 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 < 22.6% 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 ≥ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.2.2 and 3.3.2.1, and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

5. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power-High scram setpoints used in the Automated BSP Scram Region and the BSP Boundary for Specification 3.3.1.1.

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

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

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

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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07~~1.15 for two recirculation loop operation or ≥ 1.15 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 Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

~~(continued)~~

~~2.0 SLs~~

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 ≥ 22.6% 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 < 22.6% 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 ≥ 22.6% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2 AND Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.2.2 and 3.3.2.1, and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

5. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power-High scram setpoints used in the Automated BSP Scram Region and the BSP Boundary for Specification 3.3.1.1.

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

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

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

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

ATTACHMENT 3a
Revised Technical Specifications Pages for
Clinton Power Station, Unit 1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 21.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System 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 and the corporate executive responsible for overall plant nuclear safety.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),

2. LCO 3.2.2, Minimum Critical Power Ratio (MCPR) and $MCPR_{99.9\%}$,

3. LCO 3.2.3, Linear Heat Generation Rate (LHGR),

4. LCO 3.3.1.1, RPS Instrumentation (SR 3.3.1.1.14),

5. LCO 3.3.1.3, Oscillation Power Range Monitor (OPRM) Instrumentation, and

6. LCO 3.7.6, Main Turbine Bypass System, (cycle dependent thermal power limits for an inoperable Main Turbine Bypass System).

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in

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

(2) NEDO-32465, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications."

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

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

(continued)

ATTACHMENT 3b
Revised Technical Specifications Pages for
James A. FitzPatrick Nuclear Power Plant, Unit 1

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements (continued)

5.6.4 Not Used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) of Specification 3.2.1;
2. The MINIMUM CRITICAL POWER RATIO (MCPR), and MCPR(99.9%) of Specification 3.2.2;
3. The LINEAR HEAT GENERATION RATE (LHGR) of Specification 3.2.3;
4. The Reactor Protection System (RPS) APRM Neutron Flux - High (Flow Biased) Function Allowable Value of Table 3.3.1.1·1;
5. The Rod Block Monitor - Upscale Function Allowable Value of Table 3.3.2.1- 1; and
6. The Power/Flow Exclusion Region of Specification 3.4.1.

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

1. NEDE- 24011-P·A, General Electric Standard Application for Reactor Fuel;
2. NEDC-3 1317P, James A. FitzPatrick Nuclear Power Plant SAFER/GESTR- LOCA Loss-of-Coolant Accident Analysis: and
3. NED0-31960-A, BWR Owners ' Group Long-Term Stability Solutions Licensing Methodology.

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

(continued)

ATTACHMENT 3c
Revised Technical Specifications Pages for
LaSalle County Station, Units 1 and 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

(ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6.3 Radioactive Effluent Release Report

------------------------------NOTE------------------------------
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 (Deleted)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   1. The APLHGR for Specification 3.2.1.

   2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2.

   3. The LHGR for Specification 3.2.3.

(continued)

ATTACHMENT 3d
Revised Technical Specifications Pages for
Limerick Generating Station, Units 1 and 2

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with the reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 700 psia or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 and the reactor vessel steam dome pressure greater than 700 psia and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATION CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than the limit specified in the CORE OPERATING LIMITS REPORT (COLR), or THERMAL POWER greater than or equal to 90% of rated with MCPR less than the limit specified in the COLR.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER and less than 90% of RATED THERMAL POWER with MCPR less than the limit specified in the CORE OPERATING LIMITS REPORT (COLR), or THERMAL POWER greater than or equal to 90% of rated with MCPR less than the limit specified in the COLR.

ACTION:

a. With one RBM channel inoperable:

   1. Verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN, and

   2. Restore the inoperable RBM channel to OPERABLE status within 24 hours.

   Otherwise, place the inoperable rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1 and at the frequencies specified in the Surveillance Frequency Control Program unless otherwise noted in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factors for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.1.4.3,

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

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

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),*

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

---

* For Cycle 8, specific documents were approved in the Safety Evaluation dated (5/4/98) to support License Amendment No. (127).

CORE OPERATING LIMITS REPORT

6.9.1.9 Core Operating Limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the CORE OPERATING LIMITS REPORT for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1,

b. MAPFAC(P) and MAPFAC(F) factors for Specification 3.2.1,

c. The MINIMUM CRITICAL POWER RATIO (MCPR) and MCPR(99.9%) for Specification 3.2.3,

d. The MCPR(P) and MCPR(F) adjustment factor for specification 3.2.3,

e. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4,

f. The power biased Rod Block Monitor setpoints of Specification 3.3.6 and the Rod Block Monitor MCPR OPERABILITY limits of Specification 3.1.4.3.

g. The Reactor Coolant System Recirculation Flow upscale trip setpoint and allowable value for Specification 3.3.6,

h. The Oscillation Power Range Monitor (OPRM) period based detection algorithm (PBDA) setpoints for Specification 2.2.1,

i. The minimum required number of operable main turbine bypass valves for Specification 3.7.8 and the TURBINE BYPASS SYSTEM RESPONSE TIME for Specification 4.7.8.c.

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

a. NEDE-24011-P-A "General Electric Standard Application for Reactor Fuel" (Latest approved revision),

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

ATTACHMENT 3e
Revised Technical Specifications Pages for
Nine Mile Point Nuclear Station, Unit 2

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. The APLHGR for Specification 3.2.1.

2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

4. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power - High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

5. The Allowable Values, NTSPs, and MCPR conditions for the Rod Block Monitor – Upscale Functions for Specification 3.3.2.1.

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

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, (NRC approved version specified in the COLR).

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

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

(continued)

ATTACHMENT 3f
Revised Technical Specifications Pages for
Peach Bottom Atomic Power Station, Units 2 and 3

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.2.2 and 3.3.2.1, and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

5. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power-High scram setpoints used in the Automated BSP Scram Region and the BSP Boundary for Specification 3.3.1.1.

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

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

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

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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 700 psia or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 22.6% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 700 psia and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio (MCPR) for Specifications 3.2.2 and 3.3.2.1, and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Control Rod Block Instrumentation for Specification 3.3.2.1; and

5. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power-High scram setpoints used in the Automated BSP Scram Region and the BSP Boundary for Specification 3.3.1.1.

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

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

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

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

ATTACHMENT 4a
Proposed Technical Specifications Bases Changes (Mark-Up) for
Clinton Power Station, Unit 1
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur. ~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could 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. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$.~~The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR SL 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 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 ≥ 700 psia 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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 > 41.7% RTP. Thus, a THERMAL POWER limit of 21.6% RTP for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 3.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ration (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCP_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric critical power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

BASES

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

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 Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs), and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the USAR, Chapters 4, 6, and 15, and References 2, 3, 4, 5, and 6. To ensure that the MCPR Safety Limit (SL) is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is combined with~~added to~~ the MCPR$_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MCPR$_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the MCPR$_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the MCPR$_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the MCPR$_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power state (MCPR$_f$ and MCPR$_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 3, 4, and 5).

Flow dependent MCPR limits (MCPR$_f$) are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 7) and the multichannel thermal hydraulic code. MCPR$_f$ curves are provided based on the maximum credible flow runout transient. The result of a single failure or single operator error is the runout of only one loop because both recirculation loops are under independent control.

Power dependent MCPR limits (MCPR$_p$) are determined by approved transient analysis models~~the three dimensional BWR simulator code and the one dimensional transient code~~ (Ref. 8). 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 scram trips are bypassed, high and low flow MCPR$_p$ operating limits are provided for operating between 21.6% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR (MCPR$_{99.9\%}$ value, MCPR$_f$ values, and MCPR$_p$ values) are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are determined by the larger of the MCPR$_f$ and MCPR$_p$ limits, which are based on the MCPR$_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 21.6% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 21.6% RTP is unnecessary due to the large inherent margin that ensures

that the MCPR SL is not exceeded even if a limiting transient occurs.

(continued)

BASES

APPLICABILITY (continued)

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 21.6% 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 (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 21.6% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limit, 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 limit(s) 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 limit 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 the required limit 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 < 21.6% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 21.6% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 21.6% RTP and periodically 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 12 hour allowance after THERMAL POWER reaches ≥ 21.6% RTP is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

With regard to MCPR values obtained pursuant to this SR, as determined from plant indication instrumentation, the specified limit is considered to be a nominal value and therefore does not require compensation for instrument indication uncertainties (Ref. 9).

SR 3.2.2.2

Because the transient analyses may take credit for conservatism in the control rod scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analyses. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits are contained in the COLR. This determination must be preformed and any necessary changes must be implemented with in 72 hours after each set of control rod scram time tests required by SR 3.1.4.1, SR 3.1.4,2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

(continued)

BASES (continued)

REFERENCES

1. NUREG-0562, "Fuel Rod Failures As A Consequence of Nucleate Boiling or Dryout," June 1979.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).

3. USAR, Section 15.0.

4. USAR, Appendix 15B.

5. USAR, Appendix 15C.

6. NEDC-31546-P, "Maximum Extended Operating Domain and Feedwater Heater Out-of-Service Analysis for Clinton Power Station," August 1988.

7. NEDE-30130-P-A, "Steady State Nuclear Methods," April 1985.

8. NEDO-24154-A, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," General Electric Company, August 1986.

9. Calculation IP-0-0002.

ATTACHMENT 4b
Proposed Technical Specifications Bases Changes (Mark-Up) for
James A. FitzPatrick Nuclear Power Plant, Unit 1
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)
B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

JAFNPP design criteria (Ref. 1) requires. and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., 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 during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as SLMCPR(95/95), which corresponds to a 95% probability at 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.

Operation above the boundary of the nucleate boiling regime could 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

(continued)

BASES

BACKGROUND (continued)	to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of fission products to the reactor coolant.

APPLICABLE SAFETY ANALYSIS

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to 95% probability at a 95% confidence level that transition boiling will not occur, referred to as SLMCPR(95/95).~~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 limit.

2.1.1.1 Fuel Cladding Integrity

The GEXL17 critical power correlation is applicable for all critical power calculations at pressure $\geq$ 685 psig and core flows $\geq$10% of rated flow (References 5 and 6). 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 **(including the GEXL17 correlation lower limit of 685 psig)** is conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as MCPR(95/95).

The SL is based on GNF2 fuel. For cores loaded with a single fuel product line, the SLMCPR(95/95) is the MCPR(95/95) for the fuel type. For cores loaded with a mix of applicable fuel types, the SLMCPR(95/95) is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

2.1.1.3 Reactor Vessel Water Level

The reactor vessel water level is required to be above the top of the active irradiated fuel. The top of the active irradiated fuel is the top of a 150 inch fuel column which includes both the enriched and the natural uranium. During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active irradiated fuel to

provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKROUND

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 Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients, and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, 9 and 11. To ensure that ~~99.9% of the fuel rods avoid boiling transition~~the MCPR Safety Limit (SL) is not exceeded during any transient that occurs with moderate frequency, limiting transients are analyzed either with TRACG or other NRC-approved methodologies. The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature increase. The TRACG methodology calculates an operating limit MCPR (OLMCPR) for the transient initial condition that will result in no more than 0.1% of the fuel rods susceptible to boiling transition. The other methodologies calculate a reduction in CPR for each transient, with the largest change in CPR (delta-CPR) resulting from the limiting transient. When the largest delta-CPR is ~~added to~~combined with the MCPR(99.9%) ~~SL~~, an OLMCPR is obtained. The OLMCPR, calculated by either the TRACG or other methodology, sets the core operating limits.

MCPR(99.9%) is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the MCPR(99.9%) calculation are given in

Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the MCPR(99.9%) statistical analysis.

The MCPR operating limits are derived from the MCPR(99.9%) value and the transient analysis, and are dependent on the operating core flow and core exposure to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). A generator load

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

reject without bypass and a feedwater controller transient normally result in the worst case MCPR transients for a given fuel cycle.

Flow-dependent MCPR limits, MCPR(F), are necessary to assure that the MCPR SL is not violated during recirculation flow increase events. The design basis flow increase event is a slow-flow power increase event which is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. Flow runout events were analyzed along a constant xenon flow control line assuming a quasi steady-state plant heat balance. The MCPR(F) limit is specified as an absolute value and is generic and cycle-independent. The operating limit is dependent on the maximum setting of the scoop tube in the Recirculation Flow Control System.

Above the power at which the scram is bypassed (P_{Bypass}), bounding power-dependent trend functions have been developed. This trend function, Kp, is used as a multiplier to the rated MCPR operating limits to obtain the power-dependent MCPR limits, MCPR(P). Below the power at which the scram is automatically bypassed (Below P_{Bypass}), the MCPR(P) limits are actual absolute operating limit MCPR values, rather than multipliers on the rated power operating limit MCPR.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 10).

LCO

The MCPR operating limits specified in the COLR (MCPR(99.9%) value, MCPR(f) values, and MCPR(p) values) are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is a function of exposure, control rod scram times and core flow. The MCPR values for each fuel assembly must remain above the operating limit MCPR. The operating limit MCPR is determined by the larger of the MCPR(F) and MCPR(P) limits, which are based on the MCPR(99.9%) limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from the analyses of transients 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.

(continued)

ATTACHMENT 4c
Proposed Technical Specifications Bases Changes (Mark-Up) for
LaSalle County Station, Units 1 and 2
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

(continued)

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could 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. 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 instrumentation setpoints higher than this SL provides margin such that the SL 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 Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~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 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.

Cores with fuel that is all from one vendor utilize that vendor's critical power correlation for determination of MCPR. For cores with fuel from more than one vendor, the MCPR is calculated for all fuel in the core using the licensed critical power correlations. This may be accomplished by using each vendor's correlation for the vendor's respective fuel. Alternatively, a single correlation can be used for all fuel in the core. For fuel that has not been manufactured by the vendor supplying the critical power correlation, the input parameters to the reload vendor's correlation are adjusted using benchmarking data to yield conservative results compared with the critical power correlation results from the co-resident

fuel.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 700 psia 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 (approximately a mass velocity of 0.25×10^6 lb/hr-ft^2), 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 critical power 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 < 700 psia is conservative. Additional information on low flow conditions is available in References 3 and 4. Compatible ATRIUM-10 information is documented in Reference 5.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ration (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the

fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle. ~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power.~~

(continued)

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

~~Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

BASES (continued)

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	<u>2.2</u> Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved revision).
3. General Electric Services Information Letter (SIL) No. 516, Supplement 2, January 19, 1996.
4. Letter from W. C. Cline (Global Nuclear Fuel) to H. Youssefnia (Exelon Generation Company), "Clarification of SIL 516 S2 Recommendations Related to Technical Specifications for Low Pressure Conditions - Revision 1," dated September 16, 2016.
5. NEDC-33106P-A, "GEXL97 Correlation Applicable to ATRIUM-10 Fuel," Revision 2, June 2004.
6. 10 CFR 50.67.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

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 Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs), and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR Safety Limit (SL) is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is combined with~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the $MCPR_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the MCPR99.9% statistical analysis.

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in the UFSAR, Chapter 15 (Ref. 5).

Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 8) and the multichannel thermal hydraulic code (Ref. 9) to analyze slow flow runout transients on a cycle-specific basis. For core flows less than rated, the established MCPR operating limit is adjusted to provide protection of the MCPR SL in the event of an uncontrolled recirculation flow increase to the physical limit of the pump. Protection is provided for manual and automatic flow control by applying appropriate flow dependent MCPR operating limits. The MCPR operating limit for a given power/flow state is the greater of the rated conditions MCPR operating limit or the power dependent MCPR operating limit. For automatic flow control, in addition to protecting the MCPR SL during the flow run-up event, protection is provided by the flow dependent MCPR operating limit to prevent exceeding the rated flow MCPR operating limit during an automatic flow increase to rated core flow.

Power dependent MCPR limits ($MCPR_p$) are determined on a cycle-specific basis. These limits are established to protect the core from plant transients other than core flow increases, including pressurization and local control rod withdrawal events.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO	The MCPR operating limits specified in the COLR (MCPR$_{99.9\%}$ value, MCPR$_f$ values, and MCPR$_p$ values) are the result of the Design Basis Accident (DBA) and transient analysis. MCPR operating limits which include the effects of analyzed equipment out-of-service are also included in the COLR. The MCPR operating limits are determined by the larger of the MCPR$_f$ and MCPR$_p$ limits, which are based on the MCPR$_{99.9\%}$ limit specified in the COLR.

(continued)

BASES (continued)

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 slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies (Ref. 5) 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 (IRM) and average power range monitor (APRM) provide 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.

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 the 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

(continued)

BASES

ACTIONS

B.1 (continued)

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.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and periodically 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 Surveillance Frequency is controlled under the Surveillance Frequency Control Program. The 12 hour allowance after THERMAL POWER reaches ≥ 25% RTP is acceptable given the inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analyses may take credit for conservatism in the control rod scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analyses. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is then determined based either on the applicable limit associated with scram times of LCO 3.1.4, "Control Rod Scram Times," or the realistic scram times. The scram time dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

(continued)

BASES (continued)

REFERENCES

1. NUREG-0562, June 1979.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (as specified in Technical Specification 5.6.5).

3. UFSAR, Chapter 4.

4. UFSAR, Chapter 6.

5. UFSAR, Chapter 15.

6. EMF-94-217(NP) , Revision 1, "Boiling Water Reactor Licensing Methodology Summary," November 1995.

7. NFSR-0091, Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods, Commonwealth Edison Topical Report, (as specified in Technical Specification 5.6.5).

8. XN-NF-80-19(P)(A), Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors-Neutronic Methods for Design and Analysis, (as specified in Technical Specification 5.6.5).

9. XN-NF-80-19(P)(A), Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors-THERMEX Thermal Limits Methodology Summary Description, (as specified in Technical Specification 5.6.5).

ATTACHMENT 4d
Proposed Technical Specifications Bases Changes (Mark-Up) for
Limerick Generating Station, Units 1 and 2
(For Information Only)

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The Technical Specification Safety Limit is set generically on a fuel product Minimum Critical Power Ratio (MCPR) correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as SLMCPR(95/95). The fuel cladding integrity Safety Limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2~~more than 99.9% of the fuel rods avoid transition boiling~~. ~~Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and~~ MCPR greater than the ~~values~~ specified limit ~~for two recirculation loop operation and for single recirculation loop operation~~ represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate 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 which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as SLMCPR(95/95), which corresponds to a 95% probability at a 95% confidences level (the 95/95 MCPR criterion) that transition boiling will not occur.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below 700 psia or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. 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 greater than 4.5 psi. Analyses 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 greater than 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 of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor

pressure below 700 psia is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification Safety Limit value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as MCPR(95/95).

The Safety Limit is based on GNF2 fuel. For cores with a single fuel product line, the SLMCPR(95/95) is the MCPR(95/95) for the fuel type. For cores loaded with a mix of applicable fuel types, the SLMCPR(95/95) is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.~~

~~The analyses that demonstrate less than 0.1% of fuel rods enter transition boiling and determine the Safety Limit MCPR are performed using a statistical approach that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The analysis methods used to perform these calculations are described in Reference 1.~~

Reference:

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

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principle barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The Tech Spec Safety Limit is set generically on a fuel product Minimum Critical Power Ratio (MCPR) correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as SLMCPR(95/95). The fuel cladding integrity Safety Limit is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than the limit specified in Specification 2.1.2~~more than 99.9% of the fuel rods avoid transition boiling. Meeting the Safety Limit can be demonstrated by analysis that confirms less than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the values specified in Specification 2.1.2 for two recirculation loop operation and for single recirculation loop operation. Less than 0.1% of fuel rods in transition boiling and~~ MCPR greater than the ~~values~~ specified limit ~~for two recirculation loop operation and for single recirculation loop operation~~ represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate 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 which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as SLMCPR(95/95), which corresponds to a 95% probability at a 95% confidences level (the 95/95 MCPR criterion) that transition boiling will not occur.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the (GEXL) correlation is not valid for all critical power calculations at pressures below 700 psia or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. 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 greater than 4.5 psi. Analyses 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 greater than 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 of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 700 psia is conservative.

SAFETY LIMITS

BASES

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification Safety Limit value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as MCPR(95/95).

The Safety Limit is based on GNF2 fuel. For cores with a single fuel product line, the SLMCPR(95/95) is the MCPR(95/95) for the fuel type. For cores loaded with a mix of applicable fuel types, the SLMCPR(95/95) is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.~~

~~The analyses that demonstrate less than 0.1% of fuel rods enter transition boiling and determine the Safety Limit MCPR are performed using a statistical approach that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The analysis methods used to perform these calculations are described in Reference 1.~~

Reference:

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

INTENTIONALLY LEFT BLANK

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity ~~Safety Limit~~ MCPR(99.9%), and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that less than 0.1% of fuel rods in the core are susceptible to transition boiling or that the resulting MCPR does not decrease below the ~~Safety Limit~~operating limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

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

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a BWR system dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.

MCPR(99.9%) is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the MCPR(99.9%) calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the MCPR(99.9%) value and the transient analysis, and are dependent on the operating core flow and power state (MCPR(F), and MCPR(P), respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Ref. 6). Flow dependent MCPR limits (MCPR(F)) are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 7) to analyze slow flow runout transients.

Power dependent MCPR limits (MCPR(P)) are determined by ~~the codes used to evaluate transients as described in Reference 2~~approved transient analysis (Reference 2). 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 30% RTP.

The MCPR operating limits specified in the COLR (MCPR(99.9%) value, MCPR(F), and MCPR(P) values) are the result of the Design Basis Accident (DBA) and transient analysis. The operating limit MCPR is determined by ~~the~~the larger of the MCPR(F), and MCPR(P) limits, which are based on the MCPR(99.9%) limit specified in the COLR.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity MCPR(99.9%)~~Safety Limit MCPR~~, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady-state operating limit, it is required that less than 0.1% of fuel rods in the core are susceptible to transition boiling of that the resulting MCPR does not decrease below the ~~Safety Limit~~ operating limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

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

The evaluation of a given transient begins with the system initial parameters shown in FSAR Table 15.0-2 that are input to a BWR system dynamic behavior transient computer program. The codes used to evaluate transients are discussed in Reference 2.

MCPR(99.9%) is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the MCPR(99.9%) calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the MCPR(99.9%) value and the transient analysis, and are dependent on the operating core flow and power state (MCPR(F), and MCPR(P), respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Ref. 6). Flow dependent MCPR limits (MCPR(F)) are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 7) to analyze slow flow runout transients.

Power dependent MCPR limits (MCPR(P)) are determined by ~~the codes used to evaluate transients as described in~~approved transient analysis (Reference 2). 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 30% RTP.

The MCPR operating limits specified in the COLR (MCPR(99.9%) value, MCPR(F), and MCPR(P) values) 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, which are based on the MCPR(99.9%) limit specified in the COLR.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR(99.9%) ~~fuel cladding safety limit~~ is increased as noted in the COLR~~by Specification 2.1.2~~, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively.

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

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive internals vibration. The surveillance on differential temperatures below 30% RATED THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during the extended operation of the single recirculation loop mode.

Surveillance of recirculation loop flow, total core flow, and diffuser-to-lower plenum differential pressure is designed to detect significant degradation in jet pump performance that precedes jet pump failure. This surveillance is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and loop flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow and loop flow versus pump speed relationship must be verified.

BASES

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted if the MCPR(99.9%) ~~fuel cladding safety limit~~ is increased as noted in the COLR~~by Specification 2.1.2~~, APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively.

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

Additionally, surveillance on the pump speed of the operating recirculation loop is imposed to exclude the possibility of excessive internals vibration. The surveillance on differential temperatures below 30% RATED THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during the extended operation of the single recirculation loop mode.

Surveillance of recirculation loop flow, total core flow, and diffuser-to-lower plenum differential pressure is designed to detect significant degradation in jet pump performance that precedes jet pump failure. This surveillance is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be re-established each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgment of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow and loop flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow and loop flow versus pump speed relationship must be verified.

ATTACHMENT 4e
Proposed Technical Specifications Bases Changes (Mark-Up) for
Nine Mile Point Nuclear Station, Unit 2
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the MCPR is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during 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 could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp

(continued)

BASES

BACKGROUND (continued)

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. 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 Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~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 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 ≥ 700 psia 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 x 10^3 lb/hr. Full scale ATLAS test

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 23% RTP for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 7.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle. ~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The~~

~~probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in References 3, 4 and 6. Reference 3 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and Reference 4 also provides the nominal values of the parameters used in the MCPR SL statistical analysis.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

2.2

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 50.67 limits (Ref. 5). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. GE Service Information Letter No. 516, Supplement 2, "Core Flow Indication in the Low-Flow Region," January 19, 1996.
3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
4. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 (revision specified in the COLR).
5. 10 CFR 50.67, "Accident Source Term."
6. NEDC-33173-P-A, "Applicability of GE Methods to Expanded Operating Domains."
7. SIL No. 516 Supplement 2, January 19, 1996.
8. Clarification of SIL 516.S2 Recommendations Related to Technical Specifications for Low Pressure Conditions-003N8314 Revision 1.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

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 Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs) and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the USAR, Chapters 4, 6, 15 and Appendix A, and References 2, 3 and 6. To ensure that the MCPR Safety Limit (SL) is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is combined with ~~added to~~ the MCPR$_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the $MCPR_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis. The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in USAR, Chapter 15B. The determination of MCPR limits is discussed in Reference 6. The MCPR operating limit is the greater of either the flow dependent MCPR limit ($MCPR_f$) or the power dependent MCPR limit ($MCPR_p$). The power dependent multiplier increases at lower powers due to the feedwater controller failure transient because, for lower powers, the mismatch between runout and initial feedwater flow increases. This results in an increase in reactor subcooling and more severe changes in thermal limits during the event at offrated power. The flow dependent limit increases at lower flows due to recirculation flow increase events because, for lower flows, the difference between initial flow and maximum possible core flow increases. This results in an increase in reactor power and more severe changes in thermal limits during the event at offrated flow. The MCPR satisfies Criterion 2 of Reference 4.
LCO	The MCPR operating limits specified in the COLR ($MCPR_{99.9\%}$ value, $MCPR_f$ values, and $MCPR_p$ values) are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limits are~~is~~ determined by the larger of the $MCPR_f$ and $MCPR_p$ limits, which are based on the $MCPR_{99.9\%}$ limit specified in the COLR.
APPLICABILITY	The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% 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 documented in Reference 5 indicate that the nominal value of the initial MCPR expected at 23% 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 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2

(continued)

BASES

APPLICABILITY (continued)

occurs. When in MODE 2, the intermediate range monitor (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

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 the 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 < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% 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 ≥ 23% RTP and periodically 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 12 hour allowance after THERMAL POWER reaches ≥ 23% RTP is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

3. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR).

4. 10 CFR 50.36(c)(2)(ii).

5. "BWR/6 Generic Rod Withdrawal Error Analysis," General Electric Standard Safety Analysis Report, GESSAR – II, Appendix 15B.

6. NEDC-33286P, "Nine Mile Point Nuclear Station Unit 2 – APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," March 2007.

ATTACHMENT 4f
Proposed Technical Specifications Bases Changes (Mark-Up) for
Peach Bottom Atomic Power Station, Units 2 and 3
(For Information Only)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that 99.9% of the fuel rods avoid transition boiling. Meeting the SL can be demonstrated by analysis that confirms no more than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric (GE) Company fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur. ~~The MCPR fuel cladding integrity SL ensures that during normal operation and during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

(continued)

BASES

BACKGROUND (continued)	Operation above the boundary of the nucleate boiling regime could 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. 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 abnormal operational transients. The Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~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 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 limit. 2.1.1.1 Fuel Cladding Integrity GE critical power correlations are applicable for all critical power calculations at pressures ≥ 700 psia and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows: The pressure drop in the bypass region is essentially all elevation head with a value > 4.5 psi; therefore, the core pressure drop at low power and flows will always be > 4.5 psi. At power, the static head inside

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10^3 lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 22.6% RTP for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 4.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ration (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.
~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical~~

~~power result in an uncertainty in the value of the critical power. Therefore,~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

~~the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 1. Reference 1 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be adequately cooled as long as water level is above $^2/_3$ of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND	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 Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients, and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.
APPLICABLE SAFETY ANALYSES	The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR Safety Limit (SL) is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR (corrected for analytical uncertainties) is combined with ~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MCPR$_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the MCPR$_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the MCPR$_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the MCPR$_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power state (MCPR$_f$ and MCPR$_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 10) to analyze slow flow runout transients. The flow dependent operating limit, MCPR$_f$, is evaluated based on a single recirculation pump flow runout event (Ref. 9).

Power dependent MCPR limits (MCPR$_p$) are determined by approved transient analysis models~~the codes used to evaluate transients as described in~~ (Reference 2). 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 22.6% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR (MCRP$_{99.9\%}$ value, MCPR$_f$ values, and MCPR$_p$ values) 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, which are based on the MCPR$_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 22.6% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 22.6% RTP is unnecessary due to the large inherent margin

(continued)

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormal operational transients.

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that 99.9% of the fuel rods avoid transition boiling. Meeting the SL can be demonstrated by analysis that confirms no more than 0.1% of fuel rods in the core are susceptible to transition boiling or by demonstrating that the MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric (GE) Company fuel. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The 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, which 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 would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur. ~~The MCPR fuel cladding integrity SL ensures that during normal operation and during abnormal operational transients, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

(continued)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could 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. 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 abnormal operational transients. The Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~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 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 limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 700 psia and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

The pressure drop in the bypass region is essentially all elevation head with a value > 4.5 psi; therefore, the core pressure drop at low power and flows will always be > 4.5 psi. At power, the static head inside

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

the bundle is less than the static head in the bypass region because the addition of heat reduces the density of the water. At the same time, dynamic head loss in the bundle will be greater than in the bypass region because of two phase flow effects. Analyses show that this combination of effects causes bundle pressure drop to be nearly independent of bundle power when bundle flow is 28 X 10^3 lb/hr and bundle pressure drop is 3.5 psi. Because core pressure drop at low power and flows will always be > 4.5 psi, the bundle flow will be > 28 X 10 ^{3} lb/hr.

Full scale ATLAS test data taken at pressures from 14.7 psia (0 psig) to 800 psia (785 psig) indicate that the fuel assembly critical power with bundle flow at 28 X 10^3 lb/hr is approximately 3.35 MWt. This is equivalent to a THERMAL POWER > 50% RTP even when design peaking factors are considered. Therefore, a THERMAL POWER limit of 22.6% RTP for reactor pressure < 700 psia is conservative. Additional information on low flow conditions is available in Reference 5.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specification SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ration (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $MCPR_{95/95}$.

The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result~~

~~in an uncertainty in the value of the critical power. Therefore,~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

~~the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be adequately cooled as long as water level is above $^{2}/_{3}$ of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

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 Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during abnormal operational transients, and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experienced boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, 8, and 9. To ensure that the MCPR Safety Limit (SL) is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR (corrected for analytical uncertainties) is combined with ~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the $MCPR_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis.

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 6, 7, 8, and 9). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 10) to analyze slow flow runout transients. The flow dependent operating limit, $MCPR_f$, is evaluated based on a single recirculation pump flow runout event (Ref. 9).

Power dependent MCPR limits ($MCPR_p$) are determined by approved transient analysis models~~the codes used to evaluate transients as described in~~ (Reference 2). 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 22.6% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR ($MCRP_{99.9\%}$ value, $MCPR_f$ values, and $MCPR_p$ values) 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, which are based on the $MCPR_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 22.6% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 22.6% 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 22.6% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and

(continued)