10CFR50.90

BOSTON EDISON
Pilgrim Nuclear Power Station
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BECo 90-101
August 21, 1990

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

License DPR-35
Docket 50-293

PROPOSED TECHNICAL SPECIFICATION
CHANGE CONCERNING FUEL PARAMETERS

In accordance with 10CFR50.90, Boston Edison Company proposes the attached changes to Appendix A of Operating License DPR-35 for the Pilgrim Nuclear Power Station. The proposed changes revise the Technical Specifications to remove cycle-specific parameter limits as described in Generic Letter 88-16, add alternative requirements for fuel assemblies as described in Generic Letter 90-02, and upgrade the minimum critical power ratio (MCPR) safety limit as permitted by Amendment 14 to GESTAR-II. Changes are also included in this proposal to make the affected Technical Specifications consistent with the "Improved BWR Technical Specifications for BWR/4s."

R. G. Bird
R. G. Bird

DMV/jmm/4606

Commonwealth of Massachusetts)
County of Plymouth)

Then personally appeared before me, Ralph G. Bird, who being duly sworn, did state that he is Senior Vice President - Nuclear of Boston Edison Company and that he is duly authorized to execute and file the submittal contained herein in the name and on behalf of Boston Edison Company and that the statements in said submittal are true to the best of his knowledge and belief.

My commission expires: April 3, 1992 DATE — NOTARY PUBLIC

MARIE T. ... NOTARY

Attachments: A. Description of Proposed Changes
B. List of Affected Pages
C. Marked-up Technical Specification Pages
D. Replacement Technical Specification Pages
E. Core Operating Limits Report, Cycle 8

1 signed original and 37 copies

cc: See next page

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Attachment A to BECo 90-101

DESCRIPTION OF PROPOSED CHANGES

PROPOSED CHANGES:

These proposed changes to the Pilgrim Nuclear Power Station (PNPS) Technical Specifications are concerned with fuel parameters and accomplish the following:

1. Remove Cycle-specific Parameter Limits

   As recommended by Generic Letter 88-16, "Removal of Cycle-specific Parameter Limits from Technical Specifications," dated October 4, 1988, this change proposes the relocation of cycle-specific parameter limits from Technical Specifications to the Core Operating Limits Report (COLR). The affected cycle-specific parameter limits include the following:

   Flow-biased Average Power Range Monitor (APRM) Flux Scram Trip Setting
   Flow-biased APRM Rod Block Trip Setting
   Rod Block Monitor Trip Setting
   Average Planar Linear Heat Generation Rate (APLHGR) Limits
   Linear Heat Generation Rate (LHGR) Limit
   Minimum Critical Power Ratio (MCPR) Operating Limits
   Power/Flow Operating Map
   Fuel Design Features

   These cycle-specific parameter limits are proposed to be relocated to the COLR, that is provided in Attachment E. The relocation of these parameters to the COLR is in accordance with the agreement made between the NRC and General Electric on the implementation of Generic Letter 88-16. This agreement is documented in the letter from J.S. Charnley, General Electric, to M.W. Hodges, NRC, "Acceptance Implementation of Generic Letter 88-16," dated August 8, 1989.

   Only the actual cycle-specific parameters are relocated to the COLR, with a COLR reference in the Technical Specifications. The related surveillance requirements and action statements for exceeding the parameters remain in the Technical Specifications. Bases are provided in Technical Specifications for each parameter.

   This change is intended to reduce the burden on licensee and NRC resources in the processing of license amendments for each new fuel cycle to update cycle-specific parameter limits in the Technical Specifications. Such license amendments are developed using NRC-approved methodologies and are consistent with all applicable limits in the Final Safety Analysis Report (FSAR). Therefore, additional NRC review of the updates to the cycle-specific parameters for each new fuel cycle is not necessary. A new reporting requirement for the submittal of COLR revisions to the NRC allows continued trending of these cycle-specific limits without the necessity of prior NRC review and approval.

2. Alternative Requirements for Fuel Assemblies

As recommended by Generic Letter 90-02, "Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications," dated February 1, 1990, this change proposes to add requirements to Technical Specifications that provide flexibility for improved fuel performance. It would permit the timely removal of fuel rods that are found to be leaking during a refueling outage or are determined to be probable sources of future leakage. Specifically, this change would allow the substitution of Zircaloy-4 or stainless steel filler rods or open water channels for fuel rods in fuel assemblies if justified by cycle-specific reload analyses using an NRC-approved methodology.

There are currently no plans to replace fuel rods at PNPS because no problem currently exists with leaking fuel rods. However, this proposed change would permit the timely response to any future problems with leaking fuel rods. Applicable analyses of replacement fuel rods would use NRC-approved methodology and thus, would not require prior NRC review and approval. A new reporting requirement would require notification of the NRC in a revision to the COLR if more than 30 rods in the core, or 10 rods in any assembly are replaced per refueling. This proposed change would improve the response of the fuel performance program to future leaking fuel and result in potential reductions in future occupational radiation exposure and plant radiological releases.

3. Upgraded MCPR Safety Limit

By letter from A.C. Thadani, NRC, to J.S. Charnley, General Electric, dated December 27, 1987, the NRC approved Amendment 14 to GESTAR-II (NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel") that upgrades the MCPR safety limit for D-lattice BWR's using high enrichment fuel. The current MCPR safety limit of 1.07 was established by NEDE-24131, "Basis for 8x8 Retrofit Fuel Thermal Analysis Application," dated September 1978 and was based on fuel design characteristics typical of those used at the time. However, core reloads using high bundle R-factor General Electric fuel has resulted in increased conservatisms and has permitted the upgrade of the MCPR safety limit from 1.07 to 1.04. Consequently, this change proposes to upgrade the MCPR safety limit in the PNPS Technical Specifications from 1.07 to 1.04. Additionally, the MCPR operating limits in new COLR Table 3.3-1 would also be upgraded by the same amount (.03) to take advantage of this increased margin of safety.

4. Improved Technical Specifications

Technical Specification sections that are affected by this proposed change were revised to make them consistent with the "Improved BWR Technical Specifications for BWR/4s," contained in NEDE-31681, dated April 1989, as revised. In particular, major changes are proposed in current Technical Specification Sections 1.1/2.1 and 1.2/2.2 to adopt the format of Improved Technical Specification Section 2.0. These changes included removing redundant sections, relocating sections within Technical Specifications, and relocating sections to the COLR.

Additional changes are proposed to current Technical Specification Section 3.1.B.1 to make the action statement for maximum fraction of limiting power density (MFLPD) exceeding the fraction of rated power (FRP) consistent with Improved Technical Specifications. In particular, the action of adjusting the APRM gain is added as an alternative to adjusting the APRM scram and rod block trip setpoints.

5. Administrative Changes

The proposed changes include many editorial changes to update the Table of Contents, correct grammatical and spelling errors, correct references to the Final Safety Analysis Report (FSAR), make the Technical Specification format consistent, and add text inadvertently deleted in a previous amendment. These changes add to the clarity and readability of Technical Specifications and do not impact any margins of safety.

DETAILED DESCRIPTION OF CHANGES:

The following detailed description of changes is provided for each of the proposed changes discussed above. Attachment B contains a list of the affected Technical Specification pages. The marked-up pages of current Technical Specifications are provided in Attachment C. The proposed replacement Technical Specification pages are provided in Attachment D.

1. Remove Cycle-specific Parameter Limits

- As recommended by Generic Letter 88-16, new Definition 1.D is added for the COLR. It specifies that core operating limits for each reload cycle shall be determined in accordance with new Technical Specification 6.9.A.4.

- The APRM scram and rod block trip settings defined in current Technical Specifications 2.1.A.1.a and 2.1.B.1, respectively, are relocated to the COLR.

- The maximum APRM scram trip setting of 120% of rated thermal power in the last paragraph of current Technical Specification 2.1.A.1.a is relocated to new Footnote 15 in Technical Specification Table 3.1.1.

- The APRM rod block trip setting in refuel and startup modes in current Technical Specification 2.1.B.2 is relocated to Footnote 2 of Technical Specification Table 3.2.C-2.

- The bases for the APRM scram trip setting are relocated to the bases for Technical Specification 3.1. The reference in the bases to current Technical Specification 2.1.A.1 is revised to indicate the information has been relocated to the COLR. The bases for the APRM rod block trip settings are relocated to the bases for Technical Specification 3.2.

- Technical Specification Figure 2.1.1 is deleted because it contains a graphical representation of the APRM scram and rod block trip settings defined in the COLR. The figure contains no new information or requirements. The reference to this figure in current Bases 2.1 was also deleted.

- The reference in Technical Specification 3.1.B.1 to current Technical Specifications 2.1.A.1.a and 2.1.B.1 is revised to indicate the relocation of APRM trip setpoints to the COLR.

- The APRM scram trip setpoint definition provided in Technical Specification Table 3.1.1 is replaced with a reference in new Footnote 15 to the COLR. Footnote 14 is deleted because it contains formula definitions that have been relocated to the COLR.

- The rod block monitor trip setpoint and the reference to current Technical Specification 2.1.B in Technical Specification Table 3.2.C-2 are replaced with a reference to the COLR in new Footnote 1 of Table 3.2.C-2.

- The reference to Technical Specification Figures 3.11-1 through 3.11-7 for the applicable limiting values of APLHR in Technical Specification 3.11.A is revised to indicate their relocation to the COLR. Also, related text describing the figures and Technical Specification Figures 3.11-1 through 3.11-7 are deleted. The portion of this deleted information that is applicable to the current operating cycle is relocated to the COLR.

- The limiting value for LHGR in Technical Specification 3.11.B is replaced with a reference to the COLR to indicate its relocation.

- The text describing the MCPR operating limit in Technical Specifications 3.11.C and 4.11.C is relocated to the COLR. A reference to the COLR is added to Technical Specification 3.11.C for the MCPR operating limit values. Related Technical Specification Table 3.11-1 and Figure 3.11-8 are relocated to the COLR.

- The power/flow operating map in Technical Specification Figure 3.11-9 is relocated to the COLR. The reference to Figure 3.11-9 in Technical Specification 3.11.D is revised to indicate its relocation to the COLR.

- The bases for Technical Specification 3.11 are revised to delete redundant information, include references to the topical reports in new Technical Specification 6.9.A.4, and correct references to other technical specifications.

- Details of the reactor core design are relocated from Technical Specification 5.2 to the COLR.

- As recommended by Generic Letter 88-16, new Technical Specification 6.9.A.4 is added to list the NRC-approved analytical methods to be used to determine the core operating limits for each reload. The requirement that core operating limits meet all applicable limits of the safety analysis is also added. A new reporting requirement to submit COLR revisions to the NRC upon issuance is added.

2. <u>Alternative Requirements for Fuel Assemblies</u>

- As recommended by Generic Letter 90-02, new requirements are added to Technical Specification 5.2 to permit the substitution of Zircaloy-4 or stainless steel filler rods or open water channels for fuel rods if justified by reload analyses using NRC-approved methodology. A new reporting requirement is added to require NRC notification in a revision to the COLR if more than 30 rods in the core, or 10 rods in any assembly are replaced per refueling.

3. Upgraded MCPR Safety Limit

- The MCPR safety limit in new Technical Specification 2.1.2 is revised from 1.07 to 1.04. Note that this new specification replaces the current Technical Specification 1.1.A as part of the effort to adopt Improved Technical Specifications.

- As part of the effort to remove cycle-specific parameter limits, the MCPR operating limits in current Technical Specification Table 3.11-1 are relocated to the COLR. Note that these MCPR operating limits in new COLR Table 3.3-1 have been revised to upgrade them by the same amount (.03) as the MCPR safety limit.

4. Improved Technical Specifications

- Current Technical Specifications 1.1/2.1 and 1.2/2.2 are restructured as described below into a new Technical Specification 2.0.

  - The applicability and objective sections of current Technical Specifications 1.1/2.1 and 1.2/2.2 are deleted because they contain no requirements or new information needed for operation of PNPS.

  - The MCPR safety limit in current Technical Specification 1.1.A is relocated to new Technical Specification 2.1.2. The applicability conditions of reactor steam dome pressure and core flow are revised to be consistent with existing Technical Specification bases and stated in psig to be easily compared to plant instrumentation.

  - The core thermal power safety limit in current Technical Specification 1.1.B is relocated to new Technical Specification 2.1.1. The applicability conditions of reactor steam dome pressure and core flow are restated to be consistent with existing Technical Specification bases and stated in psig to be easily compared to plant instrumentation.

  - Current Technical Specification 1.1.C, Power Transient, is deleted because it is redundant to the requirements of 10CFR50.36(c)(1)(ii)(A) and 10CFR50.73(b)(3). In the case that reactor scram is accomplished by indirect means, 10CFR50 requires an analysis be performed to determine whether safety limits were exceeded when the direct scram signal failed to perform as expected. Thus, current Technical Specification 1.1.C makes no new requirements and may be deleted.

  - The reactor vessel water level safety limit in current Technical Specification 1.1.D is relocated to new Technical Specification 2.1.3. The actual safety limit is revised from not less than 12 inches above the top of active fuel to greater than the top of active

Attachment B to BECo 90-101

List of Affected Pages

Remove Pages	Insert Pages
i	i
ii	ii
iii	iii
1	1
6	6
7	7
8	8
8a	---
9	9
10	---
11	---
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36	36
37	37
38	38
39	39
40	40
---	40a
---	40b
---	40c
55a	55a
71	71
72	72
126	12
127	127
127A	127a

List of Affected Pages

Remove Pages	Insert Pages
145	145
146	146
205A	205a
205A-1	---
205B	205b
205B-1	---
205B-2	---
205C	205c
205C-1	---
205C-2	---
205C-3	---
205C-4	---
205C-5	---
205C-6	---
205D	205d
205E	205e
205E-1	---
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205E-6	---
205F	---
205G	---
205H	---
206m	206m
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fuel. As discussed later in the no significant hazards consideration, the decrease of 12 inches in this safety limit does not affect any FSAR Transient or Accident Analysis, or significantly reduce the margin of safety as defined in the basis for any technical specification.

- The following scram trip settings are deleted from the current Technical Specifications listed below because they are already provided in current Technical Specification Table 3.1.1.

Scram Trip Setting	Deleted Technical Specification
APRM (refuel or startup modes)	2.1.A.1.b
Intermediate Range Monitor (IR	2.1.A.1.c
Reactor Low Water Level	2.1.C
Turbine Stop Valve Closure	2.1.D
Turbine Control Valve Fast Closure	2.1.E
Condenser Low Vacuum	2.1.F
Main Steam Isolation Valve Closure	2.1.G

- The main steam isolation setpoint on main steam line low pressure in current Technical Specification 2.1.H is deleted because it is already provided in current Technical Specification Table 3.2.A.

- The core standby cooling system (CSCS) initiation setpoint for reactor low-low water level in current Technical Specification 2.1.I is deleted because it is already provided in current Technical Specification Table 3.2.B.

- The bases for current Technical Specifications 1.1/2.1 and 1.2/2.2 are rewritten, where feasible, to adopt Improved Technical Specification Bases. Some bases are relocated to the bases for current Technical Specifications 3.1, 3.2, and 3.6 to accompany the location of the respective trip settings. In some cases, bases paragraphs are deleted because appropriate bases are provided elsewhere in the document.

- The reactor steam dome pressure safety limit in current Technical Specification 1.2 is relocated to new Technical Specification 2.1.4.

- The reactor vessel high pressure scram trip setting in current Technical Specification 2.2.A is deleted because it is already provided in current Technical Specification Table 3.1.1.

- The relief/safety valve and safety valve settings in current Technical Specifications 2.2.B and 2.2.C are relocated to Technical Specification 3.6.D.1. The reference to current Technical Specification 2.2 in Technical Specification 4.6.D.1 is deleted because the safety valve setpoints are relocated to Technical Specification 3.6.D.1.

- The actions to be taken in the event a safety limit is violated are relocated from current Technical Specification 6.7 to new Technical Specification 2.2. This section is reworded to adopt the Improved Technical Specifications, but no substantial change in required actions is made.

- Current Technical Specification 3.1.B.1 is revised to include the option of adjusting the APRM gain if MFLPD is greater than FRP. This alternative action is consistent with Improved Technical Specifications.

5. Administrative Changes

- The Table of Contents is updated to accurately depict the Technical Specifications.

- Grammatical and editorial changes are included in the affected Technical Specification bases to improve readability. An FSAR reference in the bases for the relief/safety valve settings (new Technical Specification Bases 3.6.D) is corrected.

- The following editorial changes are made to Technical Specifications to correct grammar, spelling, punctuation, and format.

  - Heading for Technical Specification 3/4.1 revised.
  - Names of trip functions in Technical Specification Table 3.1.1 are made consistent with bases.
  - Heading for Technical Specification 3.6.C corrected on Page 126.
  - Reference to Specification 3.6.D.1 corrected in Specification 3.6.D.2.
  - Abbreviation spelled out in Technical Specification 3.6.D.3.
  - Format of heading corrected on Page 127.
  - Punctuation corrected in Technical Specification 3.6.G.1.
  - Heading for Technical Specifications 3.6.H and 4.6.H added to indicate the sections were previously deleted.
  - Punctuation added to Technical Specification 4.11 on Page 205a. Word added to applicability paragraph of Technical Specification 4.11 to clarify meaning.
  - Unnecessary specification numbers deleted from Technical Specification 3.11.C and 4.11.C.
  - Punctuation revised in Technical Specification 3.11.D.
  - Format of Technical Specification 6.9.A revised to increase clarity.
  - Heading for Technical Specification 6.9.B added to indicate the section was previously deleted.

- The word "delta" was added to Technical Specification 4.6.E.3 because the delta symbol was inadvertently deleted in Amendment 42 to the Technical Specifications.

SAFETY EVALUATION AND DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS:

The Code of Federal Regulations (10CFR50.91) requires licensees requesting an amendment to provide an analysis, using the standards in 10CFR50.92, that determines whether a significant hazards consideration exists. The following analyses are provided in accordance with 10CFR50.91 and 10CFR50.92 for these proposed Technical Specification changes.

1. Remove Cycle-specific Parameter Limits

   A. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the cycle-specific limits will still be determined by analyzing the same postulated events previously analyzed. The removal of the cycle-specific limits from the Technical Specifications has no influence or impact on a Design Basis Accident occurrence. Each Design Basis Transient and accident analysis previously addressed will be examined with respect to changes in the cycle dependent parameters using the NRC-approved reload design methodologies to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination, which will be performed per the requirements of 10CFR50.59, will ensure future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated. The plant will continue to operate within the limits specified in the Core Operating Limits Report (COLR) and to take the same actions when, or if, the limits are exceeded as required by the current Technical Specifications.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no physical alterations of plant configuration, changes to setpoints, or safety limits are proposed. As stated above, the removal of the cycle-specific limits does not influence, impact, nor contribute in any way to the improbability or consequences of any accident. The cycle-specific limits will be calculated using the NRC-approved methods. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when, or if, limits are exceeded.

C. The proposed changes do not involve a significant reduction in a safety margin because they do not affect any operating practices, limits, or safety-related equipment. The margin of safety presently provided by the current Technical Specifications remains unchanged. The proposed amendment still requires operation within the core limits as obtained from the NRC-approved reload design methodologies and appropriate actions to be taken if limits are violated. The development of the limits for future reloads will continue to conform to those methods described in the NRC-approved documentation. In addition, each future reload will involve a safety review to assure that operation of the plant within the cycle-specific limits will not involve a significant reduction in a margin of safety.

2. <u>Alternative Requirements for Fuel Assemblies</u>

A. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because any future modification of fuel assemblies must be justified by a cycle-specific reload analysis using an NRC-approved methodology. The reload analysis will postulate the same events previously analyzed using NRC-approved reload design methodologies to ensure the transient evaluation of the new reload core is bounded by previously accepted analyses. This examination, which will be performed per the requirements of 10CFR50.59, will ensure the modified reload core will not involve a significant increase in the probability or consequences of an accident previously evaluated. This proposed change will improve the response of the fuel performance program and result in potential reductions in future occupational radiation exposure and plant radiological releases.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because any future modification of fuel assemblies will be justified using NRC-approved methodology which will ensure conformance to existing design limits and safety analysis bases. This examination, which will be performed per the requirements of 10CFR50.59, will ensure the modified reload core will not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The proposed changes do not involve a significant reduction in a safety margin because any future modification of fuel assemblies will be justified using NRC-approved methodology per the requirements of 10CFR50.59. This examination will ensure the modification of fuel assemblies does not involve a significant reduction in a safety margin.

3. Upgraded Minimum Critical Power Ratio (MCPR) Safety Limit

A. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The NRC-approved methodology used to derive the upgraded MCPR safety limit of 1.04 applied the same criteria as that used to derive the current MCPR safety limit of 1.07. The upgraded MCPR safety limit value of 1.04 ensures fuel cladding protection equivalent to that provided with the 1.07 safety limit is maintained. In the safety evaluation for Amendment 14 to NEDE-24011-P-A (GESTAR-II), dated December 27, 1987, the NRC approved the use of the 1.04 MCPR safety limit for D-lattice BWRs subject to the following constraints: 1) the fuel has a beginning of life R-factor of greater than or equal to 1.04 and consists of fuel types P8 x 8R, BP8 x 8R, GE8 x 8E, or GE8 x 8EB, 2) the fuel is at least 2.80 weight percent U-235 bundle average enrichment, and 3) the lower enrichment bundles residing in the core have operated for at least 2 cycles. Because the Pilgrim Nuclear Power Station currently meets these constraints and will meet them in future reloads, the 1.04 MCPR safety limit provides the same degree of assurance for fuel cladding integrity as the 1.07 MCPR safety limit did for previous reload cores. Thus, the consequences of accidents previously evaluated are not significantly increased. The MCPR safety limit does not affect any physical system or equipment that could change the probability of an accident. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated.

B. Adoption of the proposed MCPR safety limit value does not affect the function of any component or system. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The use of the 1.04 MCPR safety limit reflects the utilization of current General Electric fuel designs and does provide the same margin of safety as 1.07 does with older General Electric fuel types as discussed in the previously referenced NRC safety evaluation. Because equivalent fuel cladding protection is provided with the 1.04 MCPR safety limit, the design criterion that 99.9 percent of all fuel rods do not experience boiling transition following any design basis transient is met. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

4. <u>Improved Technical Specifications</u>

A. Proposed changes are made to make selected sections of Technical Specifications consistent with the "Improved BWR Technical Specifications for BWR/4s," contained in NEDC-31681, dated April 1989, as revised. To accomplish this, Technical Specifications were relocated and redundant requirements deleted to clarify the format and improve readability. In addition, the following minor modifications were included:

1. The conditions for applicability for the MCPR and thermal power safety limits are revised to be consistent with Technical Specification bases and restated in psig to be easily compared to plant instrumentation. The result of this change is to increase the range of applicability of the MCPR safety limit (and correspondingly decrease the range of applicability for the thermal power safety limit) by reactor steam dome pressure of 0.3 psid. Specifically, the reactor steam dome pressure of 800 psia converts to 785.3 psig, which is rounded off to 785 psig and results in a difference of 0.3 psid. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Current Technical Specification 1.1.C is deleted because it is redundant to the requirements of 10CFR50.36(c)(1)(ii)(A) and 10CFR50.73(b)(3). In the case that reactor scram is accomplished by indirect means, 10CFR50 requires an analysis be performed to determine whether safety limits were exceeded when the direct scram signal failed to perform as expected. Thus, current Technical Specification 1.1.C makes no new requirements and may be deleted.

3. The reactor vessel water level safety limit is revised from not less than 12 inches above the top of active fuel to greater than the top of active fuel. No safety analyses or design basis transients rely on a reactor vessel water level safety limit of 12 inches above the top of active fuel. In addition, the change does not alter the automatic or manual response of the operators or plant equipment to any design basis transient. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

4. An alternative action statement is added to Technical Specification 3.1.B.1 in the event that the maximum fraction of limiting power density (MFLPD) exceeds the fraction of rated power (FRP). Specifically, the APRM gain may be adjusted such that the APRM readings are greater than or equal to MFLPD, in lieu of adjusting the APRM scram and rod block trip setpoints. Both alternative actions result in conservative adjustments in the APRM setpoints and provide adequate protection from exceeding safety limits. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

As discussed above, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed changes do not involve any changes to plant design or configuration. They only serve to conform the Technical Specifications to "Improved BWR Technical Specifications for BWR/4s." For this reason, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. The change in the range of applicability of the MCPR and thermal power safety limits of 0.3 psi does not involve a significant reduction in a safety margin. The change in the reactor vessel water level safety limit to the top of active fuel does not involve a significant reduction in a safety margin because it maintains an adequate margin for effective action before the water level reaches two-thirds core height. No fuel damage is predicted if the water level is maintained above two-thirds core height. A reactor vessel water level safety limit of the top of active fuel is consistent with the NRC-approved "Standard Technical Specifications," NUREG-0123, Revision 3, issued Fall 1980 and the "Improved Technical Specifications." Accordingly, the proposed changes do not involve a significant reduction in a safety margin.

5. Administrative Changes

A. The proposed changes include editorial changes to update the Table of Contents, correct grammatical and spelling errors, correct a reference to the Final Safety Analysis Report (FSAR), make the Technical Specification format consistent, and add text inadvertently deleted in a previous amendment. These changes add to the clarity and readability of Technical Specifications and are considered to be entirely administrative in nature. Accordingly, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

B. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because no plant design or configuration changes are involved.

C. The proposed changes do not involve a significant reduction in a safety margin because they do not affect any operating practices, limits, or safety-related equipment.

These changes have been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule for Change

This change will be implemented within 30 days following Boston Edison's receipt of its approval by the Commission.