ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 and 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

REVISED TECHNICAL SPECIFICATION PAGES

List of Attached Pages

Unit 2	Unit 3
9	9
17	17
24	24
140a	140a
140b	140b
140c	140c
256a	256a

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation, or 1.07 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.58W + 62\% - 0.58\,\Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

2.1 BASES: FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to or above the thermal power condition required by Regulatory Guide 1.49. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. Conservatism incorporated into the transient analyses is documented in References 2 and 3.

2.1 BASES: (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

3. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

3.5. BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. 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 the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated 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 transients evaluated are as described in References 7 and 10.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 and in the Supplemental Reload Licensing Analysis.

3.5.L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective actions to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limits value predominantely occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE 20566 (Draft), August 1974.

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

9. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, Supplement 1, NEDE-24081-P, November 1986.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

6.9.1 Routine Reports (Cont'd)

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"

(3) 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.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The existence of a minimum critical power ratio (MCPR) less than 1.06 for two recirculation loop operation, or 1.07 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.58W + 62\% - 0.58\Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

2.1 BASES: FUEL CLADDING INTEGRITY

The abnormal operational transients applicable to operation of the Peach Bottom Atomic Power Station Units have been analyzed throughout the spectrum of planned operating conditions up to or above the thermal power condition required by Regulatory Guide 1.49. The analyses were based upon plant operation in accordance with the operating map given in Figure 3.7.1 of the FSAR. In addition, 3293 MWt is the licensed maximum power level of each Peach Bottom Atomic Power Station Unit, and this represents the maximum steady state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable transient results as determined by the current analysis model. Conservatism incorporated into the transient analyses is documented in References 2 and 3.

2.1 BASES (Cont'd)

L. References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO 10802, February 1973.

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors", NEDO 24154 and NEDE 24154-P, Volumes I, II, and III.

3. "Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors NEDO-24154 and NEDE 24154-P, Volumes I, II, and III.

4. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

5. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

3.5 BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and References 7 and 10. 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 the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not violated 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 transients evaluated are as described in References 7 and 10.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Analysis of the abnormal operational transients is presented in References 7 and 10. Input data and operating conditions used in this analysis are shown in References 7 and 10 and in the Supplemental Reload Licensing Analysis.

3.5.L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective action to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe - TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alterations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution, for up to 43 in-core locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indication value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L. BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominantely occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss-of-Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6, 7 and 8, NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified CE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE 20566 (Draft), August 1974.

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by letter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED.

7. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December 1977.

9. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, Supplement 1, NEDE-24081-P, November 1986, and for Unit 3, NEDE-24082-P, December 1987.

10. "Methods for Performing BWR Reload Safety Evaluations," PECo-FMS-0006-A (as amended).

6.9.1 Routine Report (Cont'd)

(3) PECo-FMS-0003-A, "Steady-State Fuel Performance Methods Report"

(4) PECo-FMS-0004-A, "Methods for Performing BWR Systems Transient Analysis"

(5) PECo-FMS-0005-A, "Methods for Performing BWR Steady-State Reactor Physics Analysis"

(6) PECo-FMS-0006-A, "Methods for Performing BWR Reload Safety Evaluations"

(3) 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.

(4) The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be submitted upon issuance for each Operating Cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.