Bryce L. Shriver
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JAN 2 0 2003

U. S. Nuclear Regulatory Commission
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SUSQUEHANNA STEAM ELECTRIC STATION
SUPPLEMENT 2 TO PROPOSED AMENDMENT NO. 243 TO LICENSE NPF-14 AND PROPOSED AMENDMENT NO. 207 TO LICENSE NPF-22: ADOPTION OF NRC APPROVED GENERIC CHANGES TO IMPROVED TECHNICAL SPECIFICATIONS
PLA-5577

Docket Nos. 50-387 and 50-388

Reference: 1) *PLA-5372, R. G. Byram (PPL) to USNRC, "Proposed Amendment No 243 to License NPF-14 and Proposed Amendment No. 207 to License NPF-22: Adoption of NRC Approved Generic Changes to Improved Technical Specifications," dated October 16, 2001.*

2) *Letter from T. G. Colburn (USNRC), "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment RE: One-Time Deferral of Containment Integrated Leak Rate Test and Drywell-to-Suppression Chamber Bypass Leakage Test (TAC Nos MB2894 and MB2895)," dated March 8, 2002.*

3) *PLA-5467, B. L. Shriver (PPL) to USNRC, "Proposed Amendment No 211 to Unit 2 Licensee NPF-22: MCPR Safety Limits and Reference Changes," dated July 17, 2002.*

4) *PLA-5509, B. L. Shriver (PPL) to USNRC, "Supplement to Proposed Amendment No 243 to License NPF-14 and Proposed Amendment No 207 to License NPF-22: Adoption of NRC Approved Generic Changes to Improved Technical Specifications," dated August 23, 2002.*

The purpose of this letter is to provide revised camera-ready pages for Proposed Amendment No. 243 to License NPF-14 and Proposed Amendment No. 207 to License NPF-22: Adoption of NRC Approved Generic Changes to Improved Technical Specifications as modified by References 2 and 4. The need for the revised pages was

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discussed in a telephone call between Mr. Timothy Colburn (USNRC) and Mr. Michael Crowthers (PPL). These revised pages incorporate Amendment Nos. 202 for Unit 1 and 176 for Unit 2 (Reference 2) which were issued after issuance of Reference 1.

The attachment to this letter is a compilation of the camera-ready pages associated with References 1, 2, and 4.

The No Significant Hazards Consideration that was issued in Reference 1 is not affected by these revised camera-ready pages.

Should you have any questions or require additional information, please contact Mr. Duane L. Filchner at (610) 774-7819.

Sincerely,

B. L. Shriver

Attachments: Attachment 1 – Unit 1 Camera-Ready Pages
Attachment 2 – Unit 2 Camera-Ready Pages

copy: NRC Region I
Mr. D. J. Allard, PA DEP
Mr. T. G. Colburn, NRC Sr. Project Manager
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. Janati, DEP/BRP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

SUPPLEMENT 2 TO PROPOSED AMENDMENT NO. 243 TO LICENSE NPF-14: ADOPTION OF NRC APPROVED GENERIC CHANGES TO IMPROVED TECHNICAL SPECIFICATIONS UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment involves a revision to the Unit 1 Technical Specifications.

PPL Susquehanna, LLC
By:

B.L. Shriver
Sr. Vice President and Chief Nuclear Officer

Sworn to and subscribed before me
this 20 day of January, 2003.

Janice M. Reese
Notary Public

Notarial Seal
Janice M. Reese, Notary Public
City of Allentown, Lehigh County
My Commission Expires June 11, 2005

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-388

SUPPLEMENT 2 TO PROPOSED AMENDMENT NO. 207 TO LICENSE NPF-22: ADOPTION OF NRC APPROVED GENERIC CHANGES TO IMPROVED TECHNICAL SPECIFICATIONS UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment involves a revision to the Unit 2 Technical Specifications.

PPL Susquehanna, LLC
By:

B. L. Shriver
Sr. Vice President and Chief Nuclear Officer

Sworn to and subscribed before me
this 10th day of January, 2003.

Janice M. Reese

Notary Public

Notarial Seal
Janice M. Reese, Notary Public
City of Allentown, Lehigh County
My Commission Expires June 11, 2005

Attachment 1 to PLA-5577

Unit 1

Camera-Ready Pages

5.5 Programs and Manuals

5.5.1 (ODCM) (continued)

shall indicate the date (i.e., month and year) the change was implemented

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Cleanup, Standby Gas Treatment, Scram Discharge, Post Accident Sampling and Containment Air Monitoring Systems. The program shall include the following:

a. Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months

The provisions of SR 3.0.2 are applicable.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site to unrestricted areas, conforming to 10 CFR Part 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR Part 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5 5.9 Diesel Fuel Oil Testing Program (continued)

c. Total particulate concentration of stored fuel oil is ≤ 10 mg/liter when tested every 31 days by laboratory filtration.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program Testing Frequency.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.10b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the following exception:

a. NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the May 4, 1992 Type A test shall be performed no later than May 3, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig.

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 La for Type B and Type C tests and $\leq$ 0.75 La for Type A tests:

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq$ 0.05 La when tested at $\geq$ Pa.

   2) For each door, leakage rate is $\leq$ 5 scfh when pressurized to $\geq$ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table 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 both SSES units. The submittal shall 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 during the previous year 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 Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 COLR (continued)

(102% of 3441 MWt), remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis."

2. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc.

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

4. ANF-524(P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors."

5. ANF-1125(P)(A), "ANFB Critical Power Correlation."

6. NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.

7. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES."

8. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

9. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model.

10. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors."

11. XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors "

12. EMF-1997(P)(A), "ANFB-10 Critical Power Correlation."

13. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P.

14. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System,", Engineering Report ER-160P.

15. EMF-85-74 (P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."

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.

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5.6 Reporting Requirements

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Attachment 2 to PLA-5577

Unit 2

Camera-Ready Pages

5.5 Programs and Manuals

5.5.1 ODCM (continued)

shall indicate the date (i.e., month and year) the change was implemented.

5.5.2 Primary Coolant Sources Outside Containment

This program provides controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practicable. The systems include Core Spray, High Pressure Coolant Injection, Residual Heat Removal, Reactor Core Isolation Cooling, Reactor Water Cleanup, Standby Gas Treatment, Scram Discharge, Post Accident Sampling and Containment Air Monitoring Systems. The program shall include the following:

a Preventive maintenance and periodic visual inspection requirements; and

b. Integrated leak test requirements for each system at least once per 24 months.

The provisions of SR 3.0.2 are applicable.

5.5.3 Post Accident Sampling

This program provides controls that ensure the capability to obtain and analyze reactor coolant, radioactive gases and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

a. Training of personnel;

b. Procedures for sampling and analysis; and

c. Provisions for maintenance of sampling and analysis equipment.

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably

(continued)

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents from the site to unrestricted areas, conforming to 10 times the concentration values in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents pursuant to 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the site to unrestricted areas, conforming to 10 CFR Part 50, Appendix I;

e. Determination of cumulative dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days. Determination of projected dose contributions from radioactive effluents in accordance with the methodology in the ODCM at least every 31 days.

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2 percent of the guidelines for the annual dose or dose commitment, conforming to 10 CFR Part 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

(continued)

5.5 Programs and Manuals (continued)

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda are as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

5.5 Programs and Manuals

5.5.9 Diesel Fuel Oil Testing Program (continued)

c. Total particulate concentration of stored fuel oil is ≤ 10 mg/liter when tested every 31 days by laboratory filtration.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program Testing Frequency.

5.5.10 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.10b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross division checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and.

a. A required system redundant to system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to support system(s) for the supported systems (a) and (b) above is also inoperable.

(continued)

5.5 Programs and Manuals

5.5.11 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.12 Primary Containment Leakage Rate Testing Program

A program shall be established, implemented, and maintained to comply with the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995, as modified by the following exception:

a NEI 94-01-1995, Section 9.2.3: The first Type A test performed after the October 31, 1992 Type A test shall be performed no later than October 30, 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident, Pa, is 45.0 psig

The maximum allowable primary containment leakage rate, La, at Pa, shall be 1% of the primary containment air weight per day.

Leakage Rate Acceptance Criteria are:

a. Primary Containment leakage rate acceptance criterion is $\leq$ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 La for Type B and Type C tests and $\leq$ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq$ 0.05 La when tested at $\geq$ Pa,

   2) For each door, leakage rate is $\leq$ 5 scfh when pressurized to $\geq$ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table 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 both SSES units. The submittal shall 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 during the previous year 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 Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

(continued)

5.6 Reporting Requirements (continued)

core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents, the approved version(s) of which are specified in the COLR.

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis".

2. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors" Exxon Nuclear Company, Inc.

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

4. ANF-524(P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors".

5. ANF-1125(P)(A), "ANFB Critical Power Correlation".

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

6. NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.

7. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES".

8. ANF-89-98(P)(A) "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation.

9. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model."

10. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

11. XN-NF-79-71(P)(A) "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors."

12. EMF-1997 (P)(A) "ANFB-10 Critical Power Correlation."

13. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P.

14. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, "Engineering Report ER-160P.

15. EMF-85-74(P)(A), "RODEX 2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model."

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.

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