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ppl™

OCT 16 2001

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, DC 20555

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 243 TO LICENSE
NFP-14 AND PROPOSED AMENDMENT NO. 207 TO
LICENSE NFP-22: ADOPTION OF NRC APPROVED
GENERIC CHANGES TO IMPROVED TECHNICAL
SPECIFICATIONS
PLA-5372**

**Docket No. 50-387
and 50-388**

Pursuant to 10 CFR 50.90, PPL Susquehanna, LLC, (PPL) proposes to amend the Susquehanna Steam Electric Station Units 1 and 2 (SSES) Technical Specifications (TS). The proposed change adopts seven (7) generic changes to NUREG 1433, "Standard Technical Specifications for General Electric Plants (BWR/4)," Revision 1 (STS) that have been approved by the NRC for adoption by licensees.

The improved STS were implemented at SSES in 1998 through Amendments 178 (Unit 1) and 151 (Unit 2), using NUREG 1433, Rev. 1 as the model. The industry and the NRC staff have been working to improve the STS NUREGs, and as a result, generic changes have been developed. The proposed amendment adopts selected NRC approved generic changes to the STS NUREGs.

The proposed changes provide a significant benefit to the operation of SSES, in that they serve to maintain consistency with the STS and regulations, eliminate requirements that are duplicative of regulations, reduce excessive administrative burden, provide flexibility in administration of programs and provide necessary clarification to eliminate ambiguity in requirements.

Attachment 1 to this letter is the "Safety Assessment" supporting this change.

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Attachment 2 to this letter contains the "No Significant Hazards Considerations Evaluation" performed in accordance with the criteria of 10CFR 50.92 and the categorical exclusion for an Environmental Assessment as specified in 10CFR 51.22.

Attachment 3 to this letter contains markups of the Unit 1 and Unit 2 TS showing the proposed changes.

Attachment 4 to this letter contains the "camera ready" version of the revised Unit 1 and Unit 2 TS pages.

Attachment 5 to this letter contains, for information, markups of the associated TS Bases.

PPL requests approval of this change by June 30, 2002, and that it be made effective within 60 days of issuance to allow orderly implementation of any new or revised plant procedures or training.

If you have any questions, please contact Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

R. G. Byram

Attachments

copy: NRC Region I
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Project Manager
Mr. D. J. Allard, PA DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

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 Proposed Amendment No. 207 in support of a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

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

PPL Susquehanna, LLC
By:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 16th day of October, 2001.

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC: Docket No. 50-387

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 Proposed Amendment No. 243 in support of a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

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

PPL Susquehanna, LLC
By:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 16th day of October, 2001.

Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

Attachment 1 to PLA-5372

Safety Assessment

SAFETY ASSESSMENT

SECTION I

SUMMARY OF PROPOSED CHANGE

In accordance with 10 CFR 50.90, PPL Susquehanna, LLC (PPL) proposes to revise the Susquehanna Steam Electric Station Units 1 and 2 (SSES) Technical Specifications (TS) to incorporate seven (7) generic changes to NUREG 1433, "Standard Technical Specifications for General Electric Plants (BWR/4)," Revision 1 (STS) that have been approved by the NRC for adoption by licensees. A description of each of the seven approved generic changes follows:

TSTF-273, Revision 2: Safety Function Determination Program Clarifications

TS 5.5.11, Safety Function Determination Program," (SFDP) is revised to clarify in the requirements for the SFDP that consideration does not have to be made for a loss of power in determining loss of function. TS 5.5.11 is also revised to incorporate editorial change WOG-ED-23 for consistency in meaning. The Bases for Limiting Condition for Operation (LCO) 3.0.6 is revised to provide clarification of the "appropriate LCO for loss of function."

TSTF-273, Revision 2 is adopted with no variances.

The Bases for LCO 3.0.6 is also revised, consistent with TSTF-273, Revision 2, to document the proposed changes and to provide supporting information. The TS Bases are revised in accordance with TS 5.5.10, "TS Bases Control Program." A markup is included in this submittal for completeness.

Adoption of TSTF-273, Revision 2 has been requested for Palo Verde Units 1, 2, and 3 in a license amendment request dated April 1, 2001.

<u>TSTF-279, Revision 0: Remove "Applicable Supports" from Inservice Testing Program</u>

TS 5.5.6, "Inservice Testing Program," is revised to delete the reference to "applicable supports," which are addressed as part of the Inservice Inspection Program.

TSTF-279, Revision 0 is adopted with no variances.

TSTF-279, Revision 0 has previously been approved for incorporation in the Grand Gulf, Duane Arnold, Perry, and Monticello TS by license amendments dated June 30, 2000, October 3, 2000, May 15, 2001, and August 1, 2001, respectively.

<u>TSTF-299, Revision 0: Administrative Controls Program 5.5.2.b Test Interval and Exceptions</u>

TS 5.5.2, "Primary Coolant Sources Outside Containment," is revised to clarify the intent of refueling cycle intervals with respect to the system integrated leak test requirements (i.e., 24 month intervals), and to add a statement indicating that the provisions of Surveillance Requirement (SR) 3.0.2 are applicable to be consistent with other SRs in the LCO Sections.

TSTF-299, Revision 0 is adopted with no variances.

TSTF-299, Revision 0 has previously been approved for incorporation in the Vogtle Units 1 and 2 TS by license amendment dated May 11, 2001.

<u>TSTF-308, Revision 1: Determination of Cumulative and Projected Dose Contributions in Radioactive Effluent Controls Program</u>

TS 5.5.4, "Radioactive Effluent Controls Program," is revised to clarify the actual Generic Letter 89-01 determination requirements for cumulative and projected dose contributions.

TSTF-308, Revision 1 is adopted with no variances.

Editorial changes have been made in TS 5.5.4. The terms, "UNRESTRICTED AREAS" and "SITE BOUNDARY" have been changed to lower case, since neither term is defined in TS 1.1, "Definitions."

TSTF-348, Revision 0: Cancellation of NRC Environmental Monitoring Program with States

TS 5.6.2, "Annual Environmental Operating Report," is revised to delete the reference to collocated dosimeters in relation to the NRC TLD program, because the NRC no longer conducts the program.

TSTF-348, Revision 0 is adopted with no variances.

TSTF-363, Revision 0: Revise Topical Report References in ITS 5.6.5, Core Operating Limits Report

TS 5.6.5, "Core Operating Limits Report," (COLR) is revised to allow NRC approved Topical Reports to be identified by number and title only in the TS, and adds a requirement to provide the complete citation for each NRC approved Topical Report (i.e., number, title, revision, date, and any supplements) in the COLR.

TSTF-363, Revision 0 is adopted with no variances. To meet the specific requirements of TSTF-363, Revision 0, a sentence is added to TS 5.6.5.b, which states, "Those methods are described in the following documents, the approved version(s) of which are as specified in the COLR." The list of NRC approved Topical Reports is revised such that the reports are identified by number and title, with references to revision numbers, dates, and supplements deleted. Duplicate listings of the same Topical Reports are deleted. The COLR will be revised to reflect the complete citation for each NRC approved Topical Report.

TS 5.6.5 is also revised to add an additional document to the list of approved analytical methods. The additional reference (EMF-85-74 (P)) provides the latest approved burnup limits for Framatome – ANP fuel. This reference was previously deleted in Amendment 186 for Unit 1 and Amendment 154 for Unit 2 at the request of the NRC to streamline the list of references and only specify those references that are directly related to the COLR. The recently issued LEFM NRC SER refers to an outdated burnup limit. Based on an August 14, 2001 telecon between the NRC (R. Schaaf and Z. Abdullahi) and PPL (D. Filchner, C. Lehmann and A. Dyszel), it was concluded that the list of references in TS 5.6.5 should include the EMF-85-74 reference to provide clarification and to avoid future misunderstandings regarding the approved burnup limits.

Adoption of TSTF-363, Revision 0 has also been requested for Wolf Creek and Millstone Unit 2 in license amendment requests dated April 3, 2001 and April 11, 2001, respectively.

TSTF-364, Revision 0: Revision to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59

TS 5.5.10, "Technical Specifications (TS) Bases Control Program," is revised to reference 10 CFR 50.59 rather than "unreviewed safety question," to be consistent with the recent revisions to 10 CFR 50.59. TS 5.5.10.b is also revised to incorporate editorial change WOG-ED-24 for consistency in usage.

TSTF-364, Revision 0 is adopted with no variances.

TSTF-364, Revision 0 has previously been approved for incorporation in the Browns Ferry Units 1, 2, and 3, Hatch Units 1 and 2, Callaway, and Palisades TS by license amendments dated November 21, 2000, March 6, 2001, March 15, 2001, and July 9, 2001, respectively.

The proposed changes provide a significant benefit to the operation of SSES, in that they serve to maintain consistency with the STS and regulations, eliminate requirements that are duplicative of regulations, reduce excessive administrative burden and provide flexibility in administration of programs, provide necessary clarification to eliminate ambiguity in requirements, and eliminate unnecessary expenditure of NRC and licensee resources and ease the burden of processing license amendment requests.

SECTION II

DESCRIPTION AND BASIS (BOTH LICENSING AND DESIGN) OF THE CURRENT REQUIREMENTS

The improved STS were developed jointly by the commercial nuclear power industry, through the Nuclear Energy Institute (NEI) sponsored Technical Specification Task Force (TSTF), the reactor vendor Owners' Groups, and the NRC to standardize operational requirements and philosophies throughout the industry. PPL implemented the Improved Technical Specifications (ITS) at SSES in 1998 through Amendments 178 (Unit 1) and 151 (Unit 2), using NUREG 1433, Rev. 1 as the model.

SECTION III

EVALUATION OF PROPOSED CHANGE AND BASIS

PPL has reviewed the seven NRC approved TSTFs and has determined that the changes proposed in each TSTF, and their justification, are applicable to SSES. Whenever possible, the proposed changes are being incorporated into the TS using the same format and provisions in the NRC approved TSTFs. In some cases, due to plant specific differences or due to variations between the SSES TS and NUREG 1433, Revision 1, made during the ITS conversion process, minor modifications to the TSTFs may be necessary to properly incorporate the TSTF into the SSES TS. The summary of each proposed change includes a discussion of any variances between the TSTF, as approved by NRC, and the change as proposed by PPL. Any differences have been identified and justified. In all cases, the intent of the TSTF is maintained.

The proposed changes have been evaluated by type in accordance with 10 CFR 50.92 and found to not involve a significant hazards consideration.

SECTION IV

CONCLUSIONS

Generic changes to the STS NUREGs are part of the continuing effort to maintain and improve use of the STS. Generic changes to the STS NUREGs are proposed to the NRC by the NEI sponsored TSTF. Generic changes are prepared and reviewed using a process developed by the TSTF and the NRC to correct and improve the STS NUREGs. After approval by the NRC, generic changes are available for adoption by licensees who have implemented the ITS.

While the current SSES ITS have been implemented as a significant improvement in TS, there remains a need to continue to improve and correct the ITS as generic requirements change (e.g., due to changes in regulations, industry standards, etc). The proposed amendment consists of seven generic changes that have been approved for adoption by licensees as such improvements and corrections to the ITS.

The proposed changes have been approved by the NRC on a generic basis, and are in compliance with applicable regulations. PPL has evaluated the proposed changes for applicability to SSES, and has determined that operation of SSES in accordance with the proposed changes will not endanger the health and safety of the public.

Attachment 2 to PLA-5372

No Significant Hazards Consideration Evaluation

and

Environmental Assessment

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

PPL Susquehanna, LLC (PPL) proposes to revise the Susquehanna Steam Electric Station Units 1 and 2 (SSES) Technical Specifications (TS) to incorporate seven (7) generic changes to NUREG 1433, "Standard Technical Specifications for General Electric Plants (BWR/4)," Revision 1, that have been approved by the NRC for adoption by licensees.

The proposed changes are evaluated on the succeeding pages in groups, based upon the type of change being made, as follows:

ADMINISTRATIVE CHANGES: TSTF-273, TSTF-299, TSTF-308, TSTF-348, and TSTF-364.

LESS RESTRICTIVE CHANGES – REMOVED DETAIL: TSTF-279 and TSTF-363.

Based upon these evaluations, PPL has determined that the proposed amendment does not involve a significant hazards consideration.

10 CFR 50.92 EVALUATION
ADMINISTRATIVE CHANGES

PPL proposes to revise the SSES TS to adopt NRC approved generic changes TSTF-273, Revision 2, TSTF-299, Revision 0, TSTF-308, Revision 1, TSTF-348, Revision 0, and TSTF-364, Revision 0 to the STS as outlined in NUREG 1433, Revision 1. The proposed changes involve reformatting, renumbering, and rewording of TS with no change in intent. These changes, since they do not involve technical changes to the TS, are administrative.

An administrative change involves the movement of requirements within the current requirements, or with the modification of wording that does not affect the technical content of the current TS. These changes will also include non-technical modifications of requirements to conform to regulations or the Writer's Guide, or to provide consistency with NUREG 1433. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current TS.

In accordance with the criteria set forth in 10 CFR 50.92, PPL has evaluated these proposed TS changes and determined that they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change involves reformatting, renumbering, and rewording the existing TS. The reformatting, renumbering, and rewording process involves no technical changes to the existing TS. As such, this change is administrative in nature and does not affect the initiators of analyzed events or assumed mitigation of accidents or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. Therefore, the change does not involve a significant reduction in a margin of safety.

10 CFR 50.92 EVALUATION
LESS RESTRICTIVE CHANGES – REMOVED DETAIL

PPL proposes to revise the SSES TS to adopt NRC approved generic changes TSTF-279, Revision 0, and TSTF-363, Revision 0 to the STS as outlined in NUREG 1433, Revision 1. The proposed changes involve moving details out of the TS and into the TS Bases, the updated Final Safety Analysis Report (UFSAR), the Technical Requirements Manual (TRM) or other documents under regulatory control such as the Quality Assurance Plan. The removal of this information is considered to be less restrictive because it is no longer controlled by the TS change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG 1433 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, PPL has evaluated these proposed TS changes and determined that they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

   The proposed change relocates certain details from the TS to other documents under regulatory control. The TS Bases, UFSAR, and TRM will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the TS Bases are subject to the change control provisions in the Administrative Controls Chapter of the TS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e). Other documents are subject to controls imposed by TS or regulations. Since any changes to these documents will be evaluated, no significant increase in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no effect on any safety analyses assumptions. In addition, the details to be moved from the TS to other documents are the same as the existing TS. Since any future changes to these details will be evaluated, no significant reduction in a margin of safety will be allowed. A significant reduction in a margin of safety is not associated with the elimination of the 10 CFR 50.92 requirement for NRC review and approval of future changes to the relocated details. The proposed change is consistent with NUREG 1433, issued by the NRC staff, revising the TS to reflect the approved level of detail, which indicates that there is no significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna, LLC has evaluated the proposed changes and has determined that the proposed changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

<u>Basis</u>

1. As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The proposed changes do not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

3. There is no significant increase in individual or cumulative occupational radiation exposure. The proposed changes do not involve any physical alteration of the plant (no new or different type of equipment will be installed) or change in methods governing normal plant operation.

Attachment 3 to PLA-5372

Technical Specification Markups
(Units 1&2)

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.

<WOG-ED-23>

no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, 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)

TSTF-273 INSERT 2

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.

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.

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 ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 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.

TSTF-273
<u>INSERT 2</u>

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 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 including applicable supports. 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

5.5.1 (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

TSTF-299 INSERT 1

b. Integrated leak test requirements for each system at refueling ~~cycle intervals or less~~.

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)

TSTF-299
<u>INSERT 1</u>

least once per 24 months.

The provisions of SR 3.0.2 are applicable.

<TSTF-308 R1>

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;

unrestricted areas

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;

TSTF-308 INSERT A

e. ~~Determination of cumulative and projected 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;~~

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:

site boundary

(continued)

TSTF-308

INSERT A

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.

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. ~~The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.~~ 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)

(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:~~

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~~,~~." ~~July, 1992.~~

2. XN-NF-80-19(P)(A), ~~Volume 4, Revision 1,~~ "Exxon Nuclear Methodology for Boiling Water Reactors~~: Application of the ENC Methodology to BWR Reloads~~," Exxon Nuclear Company, Inc. ~~June 1986.~~

3. XN-NF-85-67(P)(A), ~~Revision 1,~~ "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc.~~, September 1986.~~

~~4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1 Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.~~

4 → ~~5~~. ANF-524(P)(A), ~~Revision 2 and Supplement 1, Revision 2,~~ "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors"~~,~~. ~~November 1990.~~

5 → ~~6~~. ANF-1125(P)(A) ~~and ANF-1125(P)(A), Supplement 1~~, "ANFB Critical Power Correlation"~~, April 1990~~.

6 → ~~7~~. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy~~, May 1992~~.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

7 → 8. NE-092-001A, ~~Revision 1,~~ "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company~~, December 1992 and NRC SER (November 30, 1993)~~.

~~9. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," August 1995.~~

8 → 10. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES." ~~, January 1995~~.

~~11. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation", July 1996.~~

9 → 12. ANF-89-98(P)(A) ~~Revision 1 and Revision 1 Supplement 1~~, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation~~, May 1995~~.

10 → 13. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model." ~~, January 1993~~.

11 → 14. XN-NF-80-19(P)(A), ~~Volumes 2, 2A, 2B, and 2C~~ "Exxon Nuclear Methodology for Boiling Water Reactors~~: EXEM BWR ECCS Evaluation Model~~." ~~, September 1982~~.

~~15. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.~~

12 → 16. XN-NF-79-71(P)(A) ~~Revision 2, Supplements 1, 2, and 3~~, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors." ~~March 1986~~.

(continued)

<TSTF-363 R0>

5.6 Reporting Requirements

5.6.5 COLR (continued)

13 → 17. EMF-1997(P)(A) ~~Revision 0~~, "ANFB-10 Critical Power Correlation," ~~July 1998, and EMF-1997(P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998~~.

14 → 18. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P~~, March 1999~~.

15 → 19. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, ~~Revision 0,~~ "Engineering Report ER-160P~~, May 2000~~.

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.

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

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. <WOG-ED-24>

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

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

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2. a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 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).

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TSTF-364

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

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

<WOG-ED-23>

d. Other appropriate limitations and remedial or compensatory actions.

no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s),

A loss of safety function exists when, assuming no concurrent single failure, 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)

<TSTF-273 R2>

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.

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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.

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 ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 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.

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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 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 including applicable supports. 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)

<TSTF-299 R0>

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

TSTF-299 INSERT 1

b. Integrated leak test requirements for each system at refueling cycle intervals or less.

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)

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least once per 24 months.

The provisions of SR 3.0.2 are applicable.

<TSTF-308 R1>

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;

unrestricted areas

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;

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e. ~~Determination of cumulative and projected 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;~~

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:

site boundary

(continued)

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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.

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. ~~The report shall identify the TLD results that represent collocated dosimeters in relation to the NRC TLD program and the exposure period associated with each result.~~ 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:~~ 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~~,~~" ~~July, 1992~~.

2. XN-NF-80-19(P)(A), ~~Volume 4, Revision 1~~, "Exxon Nuclear Methodology for Boiling Water Reactors~~:~~ ~~Application of the ENC Methodology to BWR Reloads~~," Exxon Nuclear Company, Inc. ~~June 1986~~.

3. XN-NF-85-67(P)(A), ~~Revision 1~~ "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company, Inc. ~~September 1986~~

~~4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1, Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.~~

4 → ~~5~~. ANF-524(P)(A), ~~Revision 2 and Supplement 1, Revision 2~~ "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors~~,~~" ~~November 1990~~.

5 → ~~6~~. ANF-1125(P)(A) ~~and ANF-1125(P)(A), Supplement 1~~, "ANFB Critical Power Correlation"~~, April 1990~~.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

6 → 7. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

7 → 8. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

~~9. NRC SER on PP&L Power Uprate LTR (November 30, 1993).~~

~~10. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," August 1995.~~

8 → 11. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES", January 1995.

9 → 12. NEDE-24011-P-A-10, "General Electric Standard Application For Reactor Fuel, February, 1991. 99

~~13. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation", July 1996.~~

10 → 14. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

11 → 15. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

12 → 16. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

~~17. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.~~

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

13 → 18. XN-NF-79-71(P)(A) ~~Revision 2, Supplements 1, 2, and 3~~, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," ~~March 1986~~.

14 → 19. EMF-1997 (P)(A) ~~Revision 0~~, "ANFB-10 Critical Power Correlation," ~~July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.~~

15 → 20. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, ~~March 1997~~.

16 → 21. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, ~~Revision 0,~~ "Engineering Report ER-160P, ~~May 2000~~.

17. EMF-85-74 (P), "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.

(continued)

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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.

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

require

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

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2. ~~a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 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).

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

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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. <WOG-ED-24>

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

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

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2. ~~a change to the updated FSAR or Bases that involves an unreviewed safety question as defined in 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).

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INSERT

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

Attachment 4 to PLA-5372

"Camera Ready" Technical Specifications

(Units 1&2)

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 and 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

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 changes 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 support 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.

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 ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests:

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa.

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 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)

(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. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy.

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

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

9. 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)

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

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

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

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

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

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

16. EMF-85-74(P), "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.

(continued)

5.6 Reporting Requirements

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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.

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 ≤ 1.0 La. During each unit startup following testing in accordance with this program, the leakage rate acceptance criteria are ≤ 0.60 La for Type B and Type C tests and ≤ 0.75 La for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 0.05 La when tested at ≥ Pa,

   2) For each door, leakage rate is ≤ 5 scfh when pressurized to ≥ 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."
6. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy.
7. NE-092-001A, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company.
8. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES."

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

9. NEDE-24011-P-A-10, "General Electric Standard Application For Reactor Fuel."

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

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

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

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

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

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

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

17. EMF-85-74P, "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 5 to PLA-5372

Technical Specification Bases Markups

(Units 1&2)

BASES

LCO 3.0.6 (continued)

entry into multiple support and supported systems' LCOs' Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.11, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

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LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to

(continued)

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BASES

LCO 3.0.6 (continued)

Conditions and Required Actions are eliminated by providing all the actions that are necessary to ensure the plant is maintained in a safe condition in the support system's Required Actions.

However, there are instances where a support system's Required Action may either direct a supported system to be declared inoperable or direct entry into Conditions and Required Actions for the supported system. This may occur immediately or after some specified delay to perform some other Required Action. Regardless of whether it is immediate or after some delay, when a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

Specification 5.5.11, "Safety Function Determination Program (SFDP)," ensures loss of safety function is detected and appropriate actions are taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other limitations, remedial actions, or compensatory actions may be identified as a result of the support system inoperability and corresponding exception to entering supported system Conditions and Required Actions. The SFDP implements the requirements of LCO 3.0.6.

Cross division checks to identify a loss of safety function for those support systems that support safety systems are required. The cross division check verifies that the supported systems of the redundant OPERABLE support system are OPERABLE, thereby ensuring safety function is retained. If this evaluation determines that a loss of safety function exists, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

TSTF-273 INSERT 1

LCO 3.0.7

There are certain special tests and operations required to be performed at various times over the life of the unit. These special tests and operations are necessary to demonstrate select unit performance characteristics, to perform special maintenance activities, and to perform

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TSTF-273

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This loss of safety function does not require the assumption of additional single failures or loss of offsite power or concurrent loss of emergency diesel generators. Since operation is being restricted in accordance with the ACTIONS of the support system, any resulting temporary loss of redundancy or single failure protection is taken into account. Similarly, the ACTIONS for inoperable offsite circuit(s) and inoperable diesel generator(s) provide the necessary restriction for cross train inoperabilities. This explicit cross train verification for inoperable AC electrical power sources also acknowledges that supported system(s) are not declared inoperable solely as a result of inoperability of a normal or emergency electrical power source (refer to the definition of OPERABILITY).

When a loss of safety function is determined to exist, and the SFDP requires entry into the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists, consideration must be given to the specific type of function affected. Where a loss of safety function is solely due to a single TS support system (e.g., loss of automatic start due to inoperable instrumentation, or loss of pump suction source due to low tank level) the appropriate LCO is the LCO for the support system. The ACTIONS for a support system LCO adequately addresses the inoperabilities of that system without reliance on entering its supported system LCO. When the loss of function is the result of multiple support systems, the appropriate LCO is the LCO for the supported system.