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FEB 29 2000

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

**SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 230 TO LICENSE
NPF-14: ALTERNATE RELOAD ANALYSIS
METHODS AND AMENDMENT NO. 193
TO LICENSE NPF-22: ALTERNATE RELOAD
ANALYSIS METHODS
PLA-5156**

**Docket Nos. 50-387
and 50-388**

The purpose of this letter is to propose revisions to the Unit 1 and Unit 2 Susquehanna Steam Electric Station (SSES) Technical Specifications for NRC approval. The revisions propose to incorporate Supplement 3 to PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements" into the Unit 1 and 2 SSES Technical Specifications Section 5.6.5, Core Operating Limits Report (COLR). Section 5.6.5 paragraph b. lists those methods that have been approved by the NRC for determination of the SSES Unit 1 and 2 core operating limits. The supplement describes alternative analysis methods to the approved analysis methodology for the analysis of the rotated bundle event, the control rod withdrawal error event, and the recirculation flow controller failure event.

Attachment 1 presents Supplement 3 to PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements". The supplement concludes that the alternative methods will result in conservative and safe core operating limits that are consistent with NRC Standard Review Plan.

Attachment 2 contains the "No Significant Hazards Considerations" and "Environmental Considerations" assessments. The "No Significant Hazards Considerations" assessment concludes that use of the proposed alternative analysis methods does not involve a significant increase in the probability or consequence of an accident previously

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evaluated; does not create the possibility of a new or different kind of accident from any accident previously evaluated; and does not involve a significant reduction in the margin of safety. The "Environmental Considerations" assessment concludes that the alternative analysis methods conform to the criteria for actions eligible for categorical exclusion as specified in 10CFR51.22(c)(9), and will not impact the environment.

Attachment 3 contains marked-up pages of the Unit 1 and Unit 2 Technical Specifications.

Attachment 4 contains "camera ready" versions of the revised Technical Specification pages.

The proposed changes have been approved by the SSES Plant Operations Review Committee and reviewed by the Susquehanna Review Committee.

PP&L requests approval of these proposed Technical Specification changes by December 10, 2000. They may be considered effective upon approval.

Please contact Mr. M. H. Crowthers at (610) 774-7766 if there are any questions concerning this submittal.

Sincerely,

R. G. Byram

Attachments

cc: NRC Region I
Mr. S. Hansell, NRC Sr. Resident Inspector
Mr. R. G. Schaaf, NRC Sr. Project Manager

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PP&L, INC. : Docket No. 50-387

PROPOSED AMENDMENT NO. 230
FACILITY OPERATING LICENSE NO. NPF-14
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 1

Licensee, PP&L, Inc., hereby files proposed Amendment No. 230 to its Facility Operating License No. NPF-14 dated July 17, 1982.

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

PP&L, INC.
BY:

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

Sworn to and subscribed before me
this 28 day of February, 2000.

Notarial Seal
Susan Grabowski-Turi, Notary Public
Allentown, Lehigh County
My Commission Expires Sept. 8, 2003

Notary Public

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PP&L, INC. : Docket No. 50-388

PROPOSED AMENDMENT NO. 193
FACILITY OPERATING LICENSE NO. NPF-22
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 2

Licensee, PP&L, Inc., hereby files proposed Amendment No. 193 to its Facility Operating License No. NPF-22 dated March 23, 1984.

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

PP&L, INC.
BY:

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

Sworn to and subscribed before me
this 28 day of February, 2000

Notarial Seal
Susan Grabowski-Turi, Notary Public
Allentown, Lehigh County
My Commission Expires Sept. 8, 2003

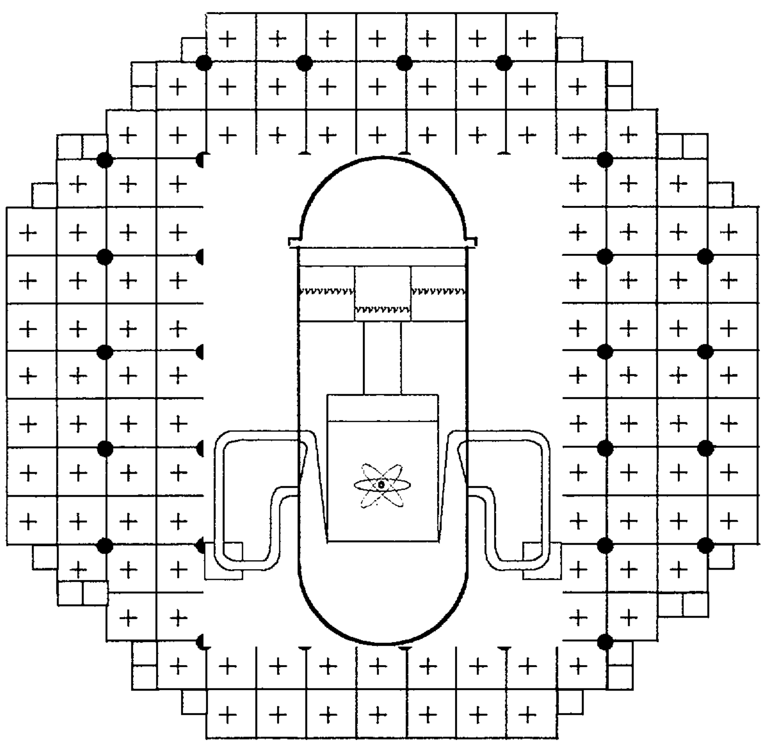
Notary Public

ATTACHMENT 1 TO PLA-5156

Supplement 3 to PL-NF-90-001, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements"

PL-NF-90-001
Supplement 3

Application of REACTOR ANALYSIS METHODS for BWR Design And Analysis



PP&L, Inc.

Application of Reactor Analysis Methods For BWR Design and Analysis

Application Enhancements

PL-NF-90-001 Supplement 3

September 1999

Principal Engineers

K. C. Knoll
C. R. Lehmann
J. M. Smith

Approved: John P Spadaro 12/22/99
J. P. Spadaro Date
Supervisor-Nuclear Fuels Engineering

12/29/99
J. M. Kulick Date
Manager-Nuclear Fuels

Legal Notice

Table of Contents

1.0 INTRODUCTION

PP&L currently performs reload licensing analysis for both units of the Susquehanna Steam Electric Station using NRC approved methods documented in References 1 through 5. These methods are used in a number of reload licensing analyses, including the analysis of events to determine the Minimum Critical Power Ratio (MCPR) operating limits. PP&L proposes to modify the current approved methodology to make better utilization of resources while continuing to provide safe and conservative operating limits for the Susquehanna Steam Electric Station. Three principal changes to PP&L's existing NRC approved methodology are being proposed:

1) Changes to the Rotated Bundle Analyses that result in methodology more consistent with that described in NUREG-0800 "Standard Review Plans" and approved for application by PP&L's current fuel vendor.

2) Revisions to the current analysis method for the Rod Withdrawal Error (RWE) event to re-incorporate credit for the Rod Block Monitor (RBM). In Supplement 2 to our approved methodology (Reference 1), commitments were made to analyze this event assuming the control rod is completely withdrawn (i.e., no credit is taken for the Rod Block Monitor). Changes to maintenance and operating procedures and Control Rod Drive related hardware have significantly reduced the possibility of a rod drift to full out event at Susquehanna, thus bringing the probability of an uncontrolled rod withdrawal to be consistent with or lower than that in the rest of the industry. Therefore, PP&L proposes to once again take credit for the Rod Block Monitor in the analysis.

3) Revisions to the Recirculation Flow Controller Failure (RFCF) analysis to use PP&L's approved steady state nodal simulation methodology. The current analysis is performed by the application of the transient analysis

code RETRAN. Due to the slowly changing conditions in the reactor core during this event, the transient modeling approach in RETRAN requires a significant amount of computer resources to model this behavior. The alternative method proposed herein is an application of PP&L's nodal analysis method (currently SIMULATE-E) to calculate the ΔCPR for this event. This approach is applicable due to the slowly changing core conditions (i.e., quasi-static methods apply). The new methodology for this event will provide accuracy comparable to the currently used RETRAN transient analysis method and also has the ability to incorporate three-dimensional effects into the analysis due to changes in void levels.

2.0 REVISION TO THE APPLICATION METHODOLOGY

As noted in the introduction, changes are being requested to three of the currently approved analysis methodologies. These include the methodologies for the Rotated Bundle, RWE and RFCF. No changes are proposed in the analysis tools as previously approved.

2.1 ROTATED BUNDLE ANALYSIS

Event Description

The rotated bundle event assumes the loading of a fuel bundle into the correct core location, but rotated by either 90 or 180 degrees from its intended orientation. The misorientation of the fuel bundle will cause the bundle to tilt toward the center of the control cell as a result of contact between the channel spacer pads and fasteners and the upper core support grid. The lack of spacer pads along the control rod face will also contribute to the lean toward the center of the control cell.

The tilted bundle geometry will cause a redistribution of water in the bypass region. This redistribution of water will cause a significant shift in the radial power distribution within the assembly. Individual fuel rods located next to the increased water gap can increase significantly in power. The assumption in the analysis is that this incorrect orientation goes unnoticed during the core loading and core verification process and the cycle operates with the misoriented bundle. The core monitoring system assumes that the assembly is oriented correctly.

Numerous preventative measures are incorporated in the fuel loading procedures to prevent the inadvertent loading of a rotated fuel bundle. SSES operation complies with GE SIL 347 (Reference 7). During insertion of any fuel bundle into the core, bundle orientation is explicitly verified. After core loading is complete, video reviews of the fully loaded core are performed in accordance with the guidance provided in GE SIL 347 and reviewed for both location and orientation, thus minimizing the possibility of a misoriented bundle during actual core operation. Scram time testing performed at the beginning of each fuel cycle also provides indication of correct bundle orientation. If a fuel bundle were misoriented, the increased friction between the control blade and the channel would result in slower blade insertion times. These slower insertion times would most likely be detected during the testing at the beginning of the cycle. As a result of these core loading and verification procedures, both units at Susquehanna have accumulated a total of 20 reactor cycles of operation without a single rotated bundle event occurring.

The current classification in the Susquehanna SES FSAR classifies the rotated bundle event as an "Infrequent Incident", which is consistent with SRP 15.4.7 (Reference 8). An infrequent incident permits some fuel damage; however, the event analysis method documented below sets a conservative criterion, which further restricts the allowable amount of cladding damage. In the proposed analysis methodology described below, the maximum amount of fuel failure will be the amount allowed for an Anticipated Operational Occurrence (AOO). Thus, the analysis methodology proposed herein will ensure that no more than 0.1% of the fuel rods in the core would experience boiling transition (i.e. potentially fail) in the event that an inadvertent rotation of a fuel bundle goes undetected during final core verification and operates in this geometry throughout the entire operating cycle. In addition, should any fuel actually fail, the radioactive release is monitored, which would therefore provide prompt indication of failed fuel.

Licensing Analysis Method

The rotated bundle analysis is performed to assure that, in the event a fuel loading error occurs during refueling operation and goes undetected during the core loading verification, the resultant radiological consequences will be a small fraction of 10 CFR 100 limits. In order to ensure these dose limits will be met, the proposed analysis methodology will ensure that at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition and the 1% plastic strain criteria. The analysis has been designed to bound the worst possible combination of rotation (i.e. 90 or 180 degrees), assembly design, assembly location and time in cycle. Effects on the rotated assembly as well as neighboring assemblies are accounted for in the analysis results. The proposed methodology, as does the NRC approved methodology currently in use by Siemens Power Corporation (Reference 6), ensures that fuel damage is limited so as not to exceed a small fraction of 10 CFR 100 dose limits.

The rotation of an assembly will cause the assembly to tilt toward the center of the control cell as a result of the spacer button and channel fastener spring contact with the upper core grid. As a result, the dimensions of the inner assembly water gap will change, decreasing the amount of bypass water adjacent to the control blade and increasing the amount of bypass water in the region away from the control blade. This displacement of bypass water causes a significant shift in the rod power distribution within the rotated and adjacent assemblies due to the changes in the water gap dimensions.

The impact of this re-distribution of water on the local peaking distributions in the rotated and surrounding assemblies is determined using PP&L's NRC approved lattice physics methodology. The displacement of the assembly is determined based on the geometric restraints in the problem (i.e. the spacer button contacting the upper core support plate). A 180° rotation is modeled since that will maximize the change in the water gap dimensions, thus maximizing the change in individual rod peaking factors.

The impact of the rotation is characterized in the ANFB-10 correlation by a change in the peaking factor parameter, F-effective. For each fuel rod, the difference between the F-effective in the rotated bundle case and the limiting F-effective for the correctly oriented case is used to determine the change in MCPR for each fuel rod in the rotated bundle. Any fuel rod that has an F-effective large enough to result in a MCPR less than the MCPR safety limit will be assumed to fail. Likewise, any fuel rod in the rotated bundle that exceeds the LHGR limit (that is based on 1% plastic strain criteria in the bundle mechanical design analysis) is also assumed to fail. If the total number of failed fuel rods exceeds 0.1%, the MCPR and/or LHGR operating limits will be adjusted until this criterion is met.

Using this methodology, typical ΔCPRs for the Rotated Bundle Analysis are approximately 0.23. This calculation, however, is cycle dependent and will be performed as part of PP&L's cycle specific licensing analyses.

2.2 ROD WITHDRAWAL ERROR ANALYSIS

Event Description

An operator erroneously selecting and continuously withdrawing a high worth control rod at its maximum withdrawal rate initiates the rod withdrawal error (RWE) event. The insertion of reactivity caused by the control rod withdrawal results in increases in both the local power around the selected control rod as well as the core average power. The event is terminated when the Rod Block Monitor (RBM) response reaches its flow-biased trip setpoint and rod movement is terminated.

Licensing Analysis Method

In Supplement 2 to PP&L's approved design and licensing methods (Reference 1), PP&L committed to not taking credit for the Rod Block Monitor. Instead, the analysis methodology conservatively assumed that the control rod withdrawal would continue until the rod was fully withdrawn. This conservative change in analysis methodology was precipitated by several rod drift events that occurred at Susquehanna. This conservatism was designed to assure conservative operating limits for the SSES units until the rod drift issue could be resolved.

The cause of the drift events was determined to be a failure of the Control Rod Drive Mechanism (CRDM) to settle back to position 00 and latch, combined with the failure of the HCU transponder card. Since that time, a number of process improvements at Susquehanna SES have corrected the conditions that led to these events. Primarily, these improvements have resulted in changes in the transponder card maintenance process and in transponder card design to improve reliability. Operations has also adopted procedures which ensure that all control rods are latched and not in the overtravel position following a scram. These improvements have reduced the probability of a control rod drift event to be consistent or better than industry experience.

Therefore, PP&L proposes to once again take credit for the RBM in the analysis of the RWE event. The plant Technical Specifications govern the actual RBM setpoint used in the analyses. Other than this modification, the analysis approach remains unchanged from that previously approved.

2.3 RECIRCULATION FLOW CONTROLLER FAILURE ANALYSIS

Event Description

The Recirculation Flow Controller Failure (RFCF) event is initiated by the failure of either the master controller or one of the individual loop controllers. This failure will cause either one (for an individual loop controller failure) or both (for a master controller failure) recirculation flow controllers to either increase or decrease the recirculation pump speed with a corresponding increase or decrease in core flow. The RFCF event with decreasing flow is not analyzed for each reload because the power decreases during the event. The decrease in power for the RFCF event with decreasing flow makes this event non-limiting from a MCPR standpoint for Susquehanna SES.

The RFCF with increasing flow is analyzed on a cycle specific basis and is a limiting event for Susquehanna SES. As the pump speed increases, the core flow and hence the reactor power will increase. The operability of the bypass valves is an important input to the event. For the bypass valve operable case, as the reactor power and steam flow increase, the bypass valves will open to regulate reactor pressure. If the reactor steam flow under these conditions exceeds the value allowed by the Maximum

Combined Flow Limiter (MCFL), pressure control will be lost, and the event will proceed similarly to the bypass valves inoperable case. If the steam flow does not exceed the MCFL, the event will be terminated when the pumps reach the run-out flow rate and a new steady state reactor power is reached.

For the bypass valve inoperable case, pressure regulation will be lost due to the inability of the control valves to accommodate the increased steam flow, and pressure and core power will increase. Under these conditions, the event is terminated when the unit scrams either on high neutron flux or high reactor pressure. Since the event may terminate on a high-pressure scram, the system's ability to remove steam is a contributor to the final peak power. Bypass valves being inoperable will increase the final peak power since pressure control will be lost.

Licensing Analysis Method

Sensitivity studies have shown that an individual loop controller failure is less limiting than the master controller failure. In addition, slower run-up rates for the master controller failure result in higher peak powers and correspondingly higher ΔCPRs. For these cases, the times from event initiation to termination for the limiting cases are long enough to allow analysis with a quasi-steady state approach using PP&L's previously approved three dimensional nodal simulation methods. A conservative set of assumptions is used to analyze the final reactor conditions to bound the transient nature of the event.

Separate analyses may be performed for both bypass valves operable and bypass valves inoperable. The plant Technical Specifications governs the operability of the bypass valves. As in the current NRC approved transient methodology, the event is initiated along the 100% equilibrium xenon rod line at various core flows. The operability state of the bypass valves as noted below governs the final conditions for the analysis.

For the bypass valve operable case, the reactor power, flow, inlet temperature and pressure at the final conditions are determined by conservatively estimating the maximum run-out rate of the recirculation pumps. The reactor inlet temperature is conservatively determined by fixing the feedwater temperature for the final conditions equal to the initial feedwater temperature. This conservatively overestimates the power increase by providing a lower core inlet temperature. The xenon concentration is fixed based on the initial conditions and not allowed to vary as the event proceeds. A power search is performed to find the new steady state power level following the event. The difference in the MCPR between the final and initial conditions of the event are used to determine the event RCPR as:

$$RCPR=(\text{Initial MCPR} - \text{Final MCPR})/ \text{Initial MCPR}$$

The event ΔCPR is then calculated for the specific cycle as:

$$\Delta CPR = MCPR\ SL * RCPR/(1 - RCPR)$$

where MCPR SL is the cycle specific MCPR safety limit.

For the bypass valves inoperable case, the steam flow will exceed the capacity of the turbine control valves. Therefore, the pressure regulation system will lose its ability to regulate the reactor pressure. At this point, the reactor pressure will increase rapidly, collapsing voids and causing a corresponding increase in core neutron power due to the increased moderation. The power and pressure rise will continue until the reactor protection system trips on either high neutron flux or high steam dome pressure.

Similar to the bypass valves operable case, the final conditions are modeled to ensure that the predicted power rise and hence change in MCPR is conservatively estimated. Sensitivity studies have shown that calculated ΔCPR from this event is maximized if the pump run-up rate is such that the trip on high reactor dome pressure and the trip on high neutron flux coincide with each other. Therefore, the core power at the final condition is set to the analytical limit for the reactor protection system trip on high neutron flux. Likewise, the core pressure used for the final condition is set to the analytical limit for the reactor protection system trip on high steam dome pressure. Due to the relatively slow nature of this event, a steady state heat balance is used to determine the inlet enthalpy. This is conservative since it overestimates the core inlet enthalpy. The core flow is determined via iterations until a critical core configuration is achieved. The resultant ΔCPR is calculated in the same manner as the bypass valves operable case.

The analysis methodology utilizing this approach yields results comparable to the currently approved RETRAN analysis. Sufficient conservatism has been incorporated to ensure the resulting analysis provides limits that would protect against violation of the MCPR safety limit in the event of an RFCF. Due to the slow nature of the event, the quasi-static approach proposed here is valid and will result in an accurate assessment of the event.

3.0 CONCLUSION

The revised applications as discussed above have been designed to provide conservative operating limits for the Rotated Bundle event, the Rod

Withdrawal Error event and the Recirculation Flow Controller Failure event. These applications are proposed as alternatives to the currently approved methodology. The design of the applications is consistent with the requirements in the NRC Standard Review Plans and assures that conservative and safe operating limits will be generated for reloads at Susquehanna.

4.0 REFERENCES

1. "Application of Reactor Analysis Methods for BWR Design and Analysis", PL-NF-90-001-A, July 1992, Supplement 1, "Loss of Feedwater Heating Changes & Use of RETRAN MOD 5.1", September 1994, and Supplement 2, "CASMO-3G Code and ANF-B Critical Power Correlation," July 1996.

2. "Qualification of Transient Analysis Methods for BWR Design and Analysis," PL-NF-87-001-A, July 1992.

3. "Qualification of Steady State Core Physics Methods for BWR Design and Analysis," PL-NF-87-001-A, July 1988.

4. "Licensing Topical Report for Power Uprate with Increased Core Flow," NE-092-001.

5. "Licensing Topical Report for Power Uprate with Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC NOS. M83426 and M83427)," Letter from Thomas E. Murley (NRC) to Robert G. Byram (PP&L), November 30, 1993.

6. “Exxon Nuclear Methodology Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads,” XN-NF-80-19(P)(A) Volume 4, Revision 1, June 1986.

7. “Misoriented Fuel Bundles,” GE SIL No. 347, December 1980.

8. “US NRC Standard Review Plan,” Section 15.4.7 “Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position,” NUREG-0800 Rev 1, July 1981.

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ATTACHMENT 2 TO PLA-5156

NO SIGNIFICANT HAZARDS CONSIDERATION
AND
ENVIRONMENTAL CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL CONSIDERATION

NO SIGNIFICANT HAZARDS CONSIDERATIONS:

PP&L Inc. proposes alternative analysis methods to the analysis methodology currently approved by the NRC for the Rotated Bundle Analysis, the Rod Withdrawal Error Analysis, and the Recirculation Flow Controller Failure Analysis. The revised methodology is proposed to be incorporated into the SSES Unit 1 and Unit 2 Technical Specification Section 5.6.5 paragraph b. This Technical Specification section lists those methods that have been approved by the NRC to be used to determine the SSES Unit 1 and 2 core operating limits.

PP&L, Inc. has evaluated the alternative analysis methodology and the associated Technical Specification revision to Section 5.6.5 paragraph b. in accordance with the criteria specified by 10CFR50.92 and has determined that the proposed alternative analysis methods do not involve a significant hazards consideration. The criteria and conclusions of our evaluation are presented below.

1. **The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.**

   This proposed alternative analysis methods do not involve an increase in the probability or consequences of an accident previously evaluated. The alternative analysis methods affect the analysis methods used to perform the Rotated Bundle Analysis, the Rod Withdrawal Error Analysis, and the Recirculation Flow Controller Failure Analysis. These events are analyzed on a cycle specific basis to ensure that the operating limits contained in the COLR will provide acceptable consequences to the health and safety of the public consistent with NRC guidelines. No physical changes are being made to plant systems, structures or components. The alternative analysis methods ensure that the off site dose consequences of the postulated events remain within the NRC guidelines.

   Based on the above, it is concluded that the alternative analysis methods do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

The alternative analysis methods do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed alternative analysis methods affect the analysis methods for the Rotated Bundle, Rod Withdrawal Error and Recirculation Flow Controller Failure Events. Since these alternative analysis methods affect analytical methods and do not affect any plant systems, structures, or components, it is concluded that the proposed alternative analysis methods do not create the possibility for any new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

The alternative analysis methods do not involve a significant reduction in the margin of safety.

The Rotated Bundle Methodology is currently analyzed as a moderate frequency event. The alternative methods will instead analyze the Rotated Bundle Event as an Infrequent Event. Analysis of this event as an infrequent event is consistent with NRC guidance (provided in the Standard Review Plan) and the frequency classification of the event as described in the SSES FSAR. The proposed analysis methodology limits the analytical off-site dose to a small fraction of 10 CFR 100 guidelines consistent with the NRC guidelines. Therefore, the proposed alternative analysis methods do not represent a significant reduction in the margin of safety.

The Rod Withdrawal Error Analysis currently does not credit the Rod Block Monitor System to limit the extent of the inadvertent rod withdrawal. The alternative proposed methods will allow credit in the analysis for the Rod Block Monitor to limit the extent of the inadvertent control rod withdrawal. Several plant and procedural improvements have been implemented that have improved the reliability of the Rod Block Monitor System. The analytical acceptance criteria for the event is not affected. Therefore, the proposed alternative analysis methods do not affect the margin of safety.

The Recirculation Flow Controller Failure analysis is currently analyzed using the RETRAN code. The proposed alternative analysis methods use PP&L's approved steady state nodal simulation methodology instead of the RETRAN code. The PP&L steady state nodal simulation methodology produces final operating limits that are consistent with the RETRAN methodology. The analytical acceptance criteria is not affected by the alternative analysis methodology. Use of PP&L's methodology does not affect the margin of safety.

ENVIRONMENTAL CONSIDERATION

An environmental assessment is not required for the proposed revisions because the requested revisions conform to the criteria for actions eligible for categorical exclusion as specified in 10CFR51.22(c)(9). The requested revisions will have no impact on the environment. As discussed above, the proposed revisions do not involve a significant hazard consideration. The proposed revisions do not involve a change in the types or increase in the amounts of effluents that may be released off-site. In addition, the proposed revisions do not involve an increase in the individual or cumulative occupational radiation exposure.

ATTACHMENT 3 TO PLA-5156

MARKED-UP TECHNICAL SPECIFICATION PAGES

5.6 Reporting Requirements

5.6.5 COLR (continued)

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.

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.

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

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.

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.

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.

18. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements." September 1999.

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

5.6.5 COLR (continued)

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.

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

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.

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.

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.

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.

20. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods For BWR Design and Analysis: Application Enhancements," September 1999.

(continued)

ATTACHMENT 4 TO PLA-5156

"CAMERA-READY" TECHNICAL SPECIFICATION PAGES

5.6 Reporting Requirements

5.6.5 COLR (continued)

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.

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.

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

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.

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.

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.

18. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements," September 1999.

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.

(continued)

5.6 Reporting Requirements

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

5.6.5 COLR (continued)

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.

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

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.

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.

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.

20. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements," September 1999.

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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