10 CFR 50.90

CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Serial: RNP-RA/00-0034

JUN 14 2000

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATION CHANGE
REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES

Ladies and Gentlemen:

This letter requests a change to the Technical Specifications (TSs) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2 in accordance with 10 CFR 50.90. The requested change proposes to revise TS 5.6.5 to incorporate analytical methodology references that are used for core operating limits. These analytical methodologies are documented in topical reports which have been accepted by the Nuclear Regulatory Commission for referencing in licensing applications.

Attachment I provides an affidavit as required by 10 CFR 50.30(b).

Attachment II provides a description of the current condition, a description of the proposed change, a safety assessment, a basis for a conclusion that the proposed change does not involve a significant hazards consideration and an environmental impact consideration which demonstrates that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9).

Attachment III provides a markup of the Current TS and Bases pages.

Attachment IV provides retyped pages for the proposed TS and Bases.

In accordance with 10 CFR 50.91(b), Carolina Power & Light (CP&L) Company is providing the State of South Carolina a copy of this letter with attachments.

A001

CP&L requests that this proposed change be reviewed and approved by August 26, 2000, in support of Cycle 21 operation, which begins at the conclusion of Refueling Outage 20, currently scheduled to begin on April 7, 2001. CP&L will implement the approved change within 30 days of approval.

If you have any questions concerning this matter, please contact Mr. H. K. Chernoff.

Sincerely,

R. L. Warden
Manager - Regulatory Affairs

ALG/alg

Attachments
- I. Affidavit
- II. Request for Technical Specification Change, Revision to Core Operating Limits Report (COLR) References
- III. Markup of Current Technical Specification and Bases Pages
- IV. Retyped Technical Specification and Bases

c: Mr. Max K. Batavia, Chief, Bureau of Radiological Health (SC)
Mr. L. A. Reyes, NRC, Region II
Mr. R. Subbaratnam, NRC NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

Affidavit

State of South Carolina
County of Darlington

J. W. Moyer, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/00-0034 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power & Light Company.

JWMoyer

Sworn to and subscribed before me

this 14th day of June 2000

(Seal) Albert L Carroll
Notary Public for South Carolina

My commission expires: March 22nd 2005

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES

Description of Current Condition

The analytical methods used to determine the core operating limits, as presented in the cycle specific Core Operating Limits Report (COLR), are currently referenced in Technical Specification (TS) 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)." There are currently 22 NRC approved methods referenced in this section.

Description of the Proposed Change

There are four specific changes proposed as follows:

1) Revise references in TS 5.6.5.b to identify the latest version as, "approved version as specified in the COLR." These references were also revised to list the topical report number first, followed by the title, then "approved version as specified in the COLR," consistent with NRC letter to Siemens Power Corporation (SPC) dated December 15, 1999.

2) Revise TS 5.6.5 to reference new Main Steam Line Break Methodology.

3) Revise TS 5.6.5 to reference new Large Break Loss of Coolant Accident (LBLOCA) Methodology.

4) Revise TS 5.6.5 to reference Generic Mechanical Design methodology.

Safety Assessment

A safety assessment of the four specific changes described above is provided below:

1) The NRC accepted the proposed method of referencing approved topical reports in Technical Specifications by letter to Siemens Power Corporation dated December 15, 1999. This method of referencing was subsequently accepted generically by NRC's approval of Technical Specification Task Force Traveler TSTF-363, Revision 0, on April 13, 2000. The proposed method of referencing topical reports would allow licensees to use current topical reports to support limits in the COLR without having to submit an amendment to the facility operating license each time a revision to the topical report is approved by the NRC. The COLR would provide specific information identifying the particular approved topical reports used to determine core limits for the particular cycle in the COLR report. This is acceptable since only NRC approved methodologies may be used, in accordance with 10 CFR 50.46, and changes to the COLR require prior licensee review for unreviewed safety questions under 10 CFR 50.59.

2) The Technical Specifications currently list XN-NF-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's," Exxon Nuclear Corporation, Richland, WA 99352, as the approved methodology for performing Main Steam Line Break Analyses. Siemens Power Corporation submitted an improvement (EMF-84-093(P)(A), Revision 1) to this methodology, and the NRC has accepted the new methodology for use in licensing applications to the extent specified and under the limitations stated in the their Safety Evaluation. The NRC Safety Evaluation was transmitted as an enclosure to NRC letter to SPC dated February 16, 1999. Carolina Power & Light (CP&L) Company has reviewed the new methodology and found the methodology acceptable for use for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2.

3) The Technical Specifications currently list the EXEM PWR Large Break LOCA Evaluation Model as accepted in letter, D. M. Crutchfield (NRC) to G. N. Ward (ENC), "Safety Evaluation of Exxon Nuclear Corporation's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports, " July 8, 1986, as the approved methodology for performing LBLOCA Analyses. There are also several sub-documents referenced in the Technical Specifications under the LBLOCA methodology as supporting references. The LBLOCA results reported in the UFSAR are based on this methodology. SPC developed a new methodology to correct problems identified within CP&L letters to the NRC dated October 14, 25, and 29, 1996. The new methodology was submitted and subsequently approved for use by the NRC by letter to Siemens Power Corporation dated June 15, 1999. CP&L has reviewed the new methodology and found the methodology acceptable for use for HBRSEP, Unit No. 2.

4) Siemens Power Corporation has established design criteria for PWR fuel in several NRC approved topical reports. SPC compiled these references into a single report in order to present to the NRC generic mechanical design criteria for SPC PWR fuel designs. The NRC approved the generic design criteria by NRC letter to Siemens Power Corporation dated February 2, 1999. CP&L has reviewed this methodology and found the methodology acceptable for use for HBRSEP, Unit 2.

No Significant Hazards Consideration Determination

The HBRSEP, Unit No. 2 TSs are proposed to be changed to replace and add analytical methodologies used to determine acceptable core designs and provide inputs to methodologies that develop the core operating limits in the Core Operating Limits Report (COLR). Carolina Power & Light (CP&L) Company has evaluated the proposed TS change and has concluded that it does not involve a significant hazards consideration. The conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed changes in a methodology have been previously generically reviewed and approved for use by the NRC for determining core operating limits. Analyzed events are assumed to be initiated by the failure of plant structures, systems, or components. The core operating limits developed in accordance with the new methodologies are bounded by the limitations in the NRC acceptance in its safety evaluations of the new methodologies. The topical reports associated with the new methodologies demonstrate that the integrity of the fuel will be maintained during normal operations and that design requirements will continue to be met. The proposed change does not have a detrimental impact on the integrity of any plant structure, system, or component. The proposed change will not alter the operation of any plant equipment, or otherwise increase its failure probability. Therefore, the probability of occurrence for a previously analyzed accident is not significantly increased.

   The consequences of a previously analyzed accident are dependent on the initial conditions assumed for the analysis, the behavior of the fuel during the analyzed accident, the availability and successful functioning of the equipment assumed to operate in response to the analyzed event, and the setpoints at which these actions are initiated. The proposed change to methodology continues to meet applicable design and safety analyses acceptance criteria. The topical reports associated with the new methodologies demonstrate that the integrity of the fuel will be maintained as is assumed or is bounded initially in accident analyses. The proposed change does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. As a result, no analyses assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. The proposed change does not affect setpoints that initiate protective or mitigative actions. The proposed change ensures that plant structures, systems, or components are maintained consistent with the safety analysis and licensing bases. Based on this evaluation, there is no significant increase in the consequences of a previously analyzed event.

   Therefore, the proposed change does not involve any increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures, or components. The proposed changes in methodology continue to meet applicable criteria for MSLB and LBLOCA analysis and assure that appropriate criteria are used in future safety analyses to establish the acceptability of reload batch fuel with regard to mechanical properties. The proposed change does not involve a physical alteration of the plant other than allowing for fuel design in accordance with NRC approved methodologies. No new or different equipment is being installed. No installed equipment is being operated in a different manner. There is no alteration to the parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigative actions. As a result no new failure modes are being introduced. There are no changes in the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The margin of safety is established through the design of the plant structures, systems, and components, through the parameters within which the plant is operated, through the establishment of the setpoints for the actuation of equipment relied upon to respond to an event, and through margins contained within the safety analyses. The proposed change in the methodologies used for MSLB and LBLOCA analyses and the use of the generic design criteria for PWR fuel designs does not impact the condition or performance of structures, systems, setpoints, and components relied upon for accident mitigation. The proposed change does not significantly impact any safety analysis assumptions or results. Therefore, the proposed change does not result in a significant reduction in the margin of safety.

Environmental Impact Consideration

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. CP&L has reviewed this request and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance with the amendment. The basis for this determination follows.

Proposed Change

The HBRSEP, Unit No. 2 Technical Specifications are changed to replace and add analytical methodologies used to determine acceptable core designs and provide inputs to methodologies that develop the core operating limits in the Core Operating Limits Report (COLR).

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons.

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

2. The proposed change is being made to change methodologies for core design, and does not involve physical changes to the facility design, configuration, operation, or maintenance. The new methodologies demonstrate by analysis and comparison of analysis with operating experience that the integrity of the fuel will be maintained during normal operations. Therefore, the proposed change does not effect actual plant effluents.

3. The proposed change does not involve physical changes to the facility design, configuration, operation, or maintenance. The new methodologies demonstrate by analysis and comparison of analysis with operating experience that the integrity of the fuel will be maintained during normal operations. Therefore the proposed change does not effect individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/00-0034
5 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT; UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES

MARKUP OF CURRENT TECHNICAL SPECIFICATION AND BASES PAGES

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Control Bank Insertion Limits for Specification 3.1.6;

5. Heat Flux Hot Channel Factor ($F_Q(Z)$) limit for Specification 3.2.1;

6. Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) limit for Specification 3.2.2;

7. Axial Flux Difference (AFD) limits for Specification 3.2.3; and

8. Boron Concentration limit for Specification 3.9.1 .

version

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved ~~revision number~~ shall be identified in the COLR. These methods are those specifically described in the following documents:

1. XN-75-27(A), ~~latest Revision and Supplements,~~ "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

2. XN-NF-84-73(P), ~~latest Revision and Supplements,~~ "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

3 . XN-NF-82-21(A), ~~latest Revision,~~ "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

approved version as specified in the COLR

Steam Line Break Methodology as defined by:

ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," approved version as specified in the COLR.

EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

4. ~~XN-NF-84-93(A), latest Revision and Supplements, "Steamline Break Methodology for PWR's," Exxon Nuclear Corporation, Richland, WA 99352~~.

5. XN-75-32(A), ~~Supplements 1, 2, 3, 4,~~ "Computational Procedure for Evaluating Rod Bow," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

6. XN-NF-82-49(A), ~~latest Revision and Supplements,~~ "Exxon Nuclear Corporation Evaluation Model EXEM PWR Small

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Break Model," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

7. ~~EXEM PWR Large Break LOCA Evaluation Model as accepted in Letter, D. M. Crutchfield (NRC) to G. N. Ward (ENC), "Safety Evaluation of Exxon Nuclear Corporation's Large Break ECCS Evaluation Model EXEM/PWR and Acceptance for Referencing of Related Licensing Topical Reports," July 8, 1986.~~

~~EXEM PWR LBLOCA Model includes the following references:~~

~~XN-NF-82-20(P), latest Revision and Supplements, "Exxon Nuclear Corporation Evaluation Model EXEM/PWR ECCS Model Updates," Exxon Nuclear Corporation, Richland, WA 99352.~~

~~XN-NF-82-07(A), latest Revision, "Exxon Nuclear Corporation ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Corporation, Richland, WA 99352.~~

~~XN-NF-81-58(A), latest Revision, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Exxon Nuclear Corporation, Richland, WA 99352.~~

~~XN-NF-85-16(P), Volume 1 and Supplements, Volume 2, latest Revision and Supplements, "PWR 17x17 Fuel Cooling Test Program," Exxon Nuclear Corporation, Richland, WA 99352.~~

~~XN-NF-85-105(P), and Supplements, "Scaling of FCTF Based Reflood Heat Transfer Correlation for Other Bundle Designs," Exxon Nuclear Corporation, Richland, WA 99352.~~

8. XN-NF-78-44(A), ~~latest Revision~~, "Generic Control Rod Ejection Analysis," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

EMF-2087 (P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

approved version as specified in the COLR

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. XN-NF-621(A), ~~latest Revision~~, "XNB Critical Heat Flux Correlation," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

10. ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," ~~Advanced Nuclear Fuels Corporation, Richland, WA 99352~~.

11. XN-NF-82-06(A), ~~latest Revisions and Supplements~~, "Qualification of Exxon Nuclear Fuel for Extended Burnup," ~~Exxon Nuclear Corporation, Richland, WA 99352~~.

approved version as specified in the COLR

12. ~~Meyer, P. E. and Kornfilt, J.~~, "NOTRUMP, A Nodal Transient Small Break and General Network Code," WCAP-10080-A, ~~August 1985~~.

13. ~~Lee, N., Tauche, W. D., Schwartz, W. R.,~~ "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," WCAP-10081-A, ~~August 1985~~.

14. ~~Bordelon, F. M., et. al., "LOCTA-IV Program: Loss of Coolant Transient Analysis,"~~ WCAP-8301 (Proprietary) and WCAP-8305 (Nonproprietary), ~~June 1974~~.

15. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power & Light Co., H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," USNRC, Washington, DC 20555, 7 Nov. 84.

16. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," ~~Advanced Nuclear Fuels Corporation, Richland, WA 99352, latest revisions and supplements~~.

17. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," ~~Advanced Nuclear Fuels Corporation, Richland, WA 99352, latest revisions and supplements~~.

18. ANF-89-151(A), ~~latest Revision and Supplements,~~ "ANF-RELAP Methodology for Pressurized Water Reactors:

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

Analysis of Non-LOCA Chapter 15 Events," ~~Advanced Nuclear Fuels Corporation, Richland WA 99352~~.

19. EMF-92-081(A), ~~latest Revision and Supplements~~, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," ~~Siemens Power Corporation - Nuclear Division, Richland, WA 99352~~.

approved version as specified in the COLR

20. EMF-92-153(P)(A), ~~Revision 0 and Supplement 1~~, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," ~~Siemens Power Corporation, Richland WA 99352, March 7, 1994~~.

21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results,"~~, Exxon Nuclear Company, November 1986~~.

22. EMF-96-029(P)(A), ~~Volume 1, Volume 2, and Attachment~~, "Reactor Analysis System for PWRs," ~~Siemens Power Corporation, Richland WA, 99352, January 1997~~.

23. EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

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 .

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

(continued)

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/00-0034
4 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATION CHANGE
<u>REVISION TO CORE OPERATING LIMITS REPORT (COLR) REFERENCES</u>

RETYPED TECHNICAL SPECIFICATION AND BASES

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Control Bank Insertion Limits for Specification 3.1.6;

5. Heat Flux Hot Channel Factor ($F_Q(Z)$) limit for Specification 3.2.1;

6. Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) limit for Specification 3.2.2;

7. Axial Flux Difference (AFD) limits for Specification 3.2.3; and

8. Boron Concentration limit for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved version shall be identified in the COLR. These methods are those specifically described in the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in COLR.

2. XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," approved version as specified in COLR.

3. XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in COLR.

4. Steam Line Break Methodology as defined by:

ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," approved version as specified in the COLR.

EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

5. XN-75-32(A), "Computational Procedure for Evaluating Rod Bow," approved version as specified in COLR.

6. XN-NF-82-49(A), "Exxon Nuclear Corporation Evaluation Model EXEM PWR Small Break Model," approved version as specified in COLR.

7. EMF-2087 (P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

8. XN-NF-78-44(A), "Generic Control Rod Ejection Analysis," approved version as specified in COLR.

9. XN-NF-621(A), "XNB Critical Heat Flux Correlation," approved version as specified in COLR.

10. ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in COLR.

11. XN-NF-82-06(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in COLR.

12. WCAP-10080-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," approved version as specified in COLR.

13. WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," approved version as specified in COLR.

14. WCAP-8301 (Proprietary) and WCAP-8305 (Nonproprietary), "LOCTA-IV Program: Loss of Coolant Transient Analysis," approved version as specified in COLR.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

15. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power & Light Co., H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," USNRC, Washington, DC 20555, 7 Nov. 84.

16. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in COLR.

17. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in COLR.

18. ANF-89-151(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in COLR.

19. EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in COLR.

20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in COLR.

21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in COLR.

22. EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in COLR.

23 EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(continued)