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10 CFR 50.90
10 CFR 50.91

102-04455-CDM/SAB/JAP
June 8, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Station P1-37
Washington, DC 20555-0001

References:

1) Letter dated March 21, 1990, "Acceptance for Referencing of Topical Report YAEC-1363, CASMO-3G Validation," from USNRC to G. Papanic, Jr., Yankee Atomic Electric Company

2) Letter dated February 20, 1990, "Acceptance for Referencing of Topical Report YAEC-1659, SIMULATE-3, Validation Verification," from USNRC to G. Papanic, Jr., Yankee Atomic Electric Company

3) Letter dated August 10, 1992, "Acceptance of Topical Report SCE-9001, PWR Reactor Physics Methodology using CASMO-3/SIMULATE-3," from USNRC to Harold B. Ray, Southern California Edison Company

4) Letter dated November 23, 1992, "Acceptance for Referencing of Topical Report DPC-NE-1004, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," from USNRC to H. B. Tucker, Duke Power Company

5) Letter dated January 3, 2000, "NSPNAD-8101, Revision 2: Prairie Island Nuclear Power Plant, Qualification of Reactor Physics Methods for Application to Prairie Island," from Joel P. Sorensen, Prairie Island Nuclear Generating Plant to USNRC

6) Letter dated December 30, 1998, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2", from James F. Mallay, Siemens Power Corporation to USNRC

APO1

Change : NRC PDR 1 ltr 1 encl w/o prop.

7) Letter dated October 18, 1999, "Acceptance for Referencing of Licensing Topical Report EMF-2158(P), Revision 0, Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," from USNRC to James F. Mallay, Siemens Power Corporation

Dear Sirs:

Subject: **Palo Verde Nuclear Generating Station (PVNGS)**
Units 1, 2 and 3
Docket Nos. STN 50-528/529/530
Request for Amendment to Technical Specification
5.6.5, Core Operating Limits Report (COLR)
(CASMO-4/SIMULATE-3)

Pursuant to 10 CFR 50.90, Arizona Public Service Company (APS) requests an amendment to Technical Specification 5.6.5, Core Operating Limits Report (COLR), for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3. This request includes NRC review and approval of the enclosed report, "Arizona Public Service Company PWR Reactor Physics Methodology using CASMO-4/SIMULATE-3, September 1999". The proposed amendment would add methodology using CASMO-4 and SIMULATE-3 codes to the list of analytical methods used to determine core operating limits contained in Technical Specification 5.6.5.b. The change will allow the use of the CASMO-4 and SIMULATE-3 methodology to perform nuclear design calculations.

The CASMO-3/SIMULATE-3 program package has been previously submitted to the NRC for review and approval. Yankee Atomic Electric Company submitted Topical Reports YAEC-1363 and YAEC-1659, Southern California Edison submitted Topical Report SCE-9001, and Duke Power submitted Topical Report DPC-NE-1004. These submittals were found to be acceptable by the NRC through issued Safety Evaluation Reports (SERs) (References 1-4). Currently, the NRC is reviewing Northern States Power Company submittal for use of CASMO-4/SIMULATE-3 (Reference 5). Additionally, Siemens Power Corporation submitted to the NRC for review and approval a topical report for use of CASMO-4 methodology (Reference 6). This report was found to be acceptable for use only by Boiling Water Reactors (BWR), as documented in the NRC SER (Reference 7).

Provided in Enclosure 1 to this letter are the following sections which support the proposed Technical Specification amendment:

A. Description of the Proposed Technical Specification Amendment
B. Purpose of the Technical Specification
C. Need for the Technical Specification Amendment
D. Safety Analysis of the Proposed Technical Specification Amendment
E. No Significant Hazards Consideration Determination
F. Environmental Consideration
G. Marked-up Technical Specification Page
H. Retyped Technical Specification Page

Provided in Enclosure 2 is the "Arizona Public Service Company PWR Reactor Physics Methodology using CASMO-4/SIMULATE-3, September 1999" report. This report documents the validation and level of accuracy of the reactor core physics method to be used by APS to perform analyses for Pressurized Water Reactors (PWR). This report also demonstrates APS' proficiency to set up input decks, execute the codes, and properly interpret the results using CASMO-4/SIMULATE-3.

Contained within "Arizona Public Service Company PWR Reactor Physics Methodology using CASMO-4/SIMULATE-3, September 1999" report (Enclosure 2), is information that ABB Combustion Engineering Nuclear Power (ABB/CE) has determined to be proprietary. APS has obtained the proprietary material from documents, which identify this information as being owned by ABB/CE and for which ABB/CE has executed affidavits, which set forth the bases on which the information may be withheld from public disclosure by the NRC. The following is a list of documents submitted to the NRC by ABB/CE, which contained the subject affidavits:

- Letter dated December 3, 1974, "LD-74-533, Topical Report CENPD-153P, Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems," from W. R. Corcoran, Combustion Engineering, Inc., to USNRC

- Letter dated December 11, 1981, "LD-81-094, Submittal of CENPD-226, The ROCS and DIT Computer Codes for Nuclear Design," from A. E. Scherer, Combustion Engineering, Inc., to USNRC

- Letter dated November 2, 1990, "LD-90-086, Topical Report CENPD-382-P, Methodology for Core Designs Containing Erbium Burnable Absorbers," from S. A. Toelle, ABB Combustion Engineering, Inc., to USNRC

- Letter dated February 25, 1992, "LD-92-032, Transmittal of Physics Methods and Performance Verification for Core Designs Containing Erbium Burnable Absorbers," from S. A. Toelle, ABB Combustion Engineering, Inc., to USNRC

- Letter dated March 11, 1999, "LD-99-016, Application of an ENDF/B-VI Based DIT Cross Section Library to Nuclear Core Design and Safety Analyses," from I. C. Rickard, ABB Combustion Engineering, Inc., to USNRC

Enclosure 2 of this submittal is appropriately annotated as "Proprietary". Enclosure 3 of this submittal is a "Redacted" version of this same report. Based on the above, we request that the proprietary information be withheld from public disclosure.

In accordance with the PVNGS Quality Assurance Program, the Plant Review Board and Offsite Safety Review Committee have reviewed and concurred with this proposed amendment. By copy of this letter this request is being forwarded to the Arizona Radiation Regulatory Agency (ARRA) pursuant to 10 CFR 50.91(b)(1).

APS requests that the enclosed Technical Specification amendment request be reviewed and approved by January 31, 2001, with an allowance of 45 days for implementation of the approved amendment.

No commitments are being made to the NRC by this letter.

Should you have any questions, please contact Scott A. Bauer at (623) 393-5978.

Sincerely,

David Mauldin

CDM/SAB/JAP/kg

Enclosures

cc: E. W. Merschoff (all w/Enclosures)
M. B. Fields
J. H. Moorman
A. V. Godwin (ARRA)

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, David Mauldin, represent that I am Vice President Nuclear Engineering and Support, Arizona Public Service Company (APS), that the foregoing document has been signed by me on behalf of APS with full authority to do so, and that to the best of my knowledge and belief, the statements made therein are true and correct.

David Mauldin

David Mauldin

Sworn To Before Me This 8th Day Of June, 2000.

Nora E. Meador

Notary Public

"OFFICIAL SEAL"
Nora E. Meador
Notary Public-Arizona
Maricopa County
My Commission Expires 4/6/2003

Notary Commission Stamp

ENCLOSURE 1

Proposed Amendment to Units 1, 2 and 3
Technical Specification 5.6.5.b

A. DESCRIPTION OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The proposed Technical Specification (TS) amendment will add an additional methodology to the list of analytical methods contained in Technical Specification 5.6.5.b. Specifically, the additional methodology to be added is:

> "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," September 1999 [Methodology for Specifications 3.1.1, Shutdown Margin – Reactor Trip Breakers Open; 3.1.2, Shutdown Margin – Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Mode 6)].

APS intends to replace ABB Combustion Engineering Nuclear Power (ABB/CE) DIT and ROCS/MC computer codes with the Studsvik CASMO-4/SIMULATE-3 computer codes while retaining the ability to use DIT/ROCS/MC. The proposed change will not alter any reload analysis methodology other than replacing the DIT/ROCS/MC code package with an equivalent code package.

APS intends to use the CASMO-4/SIMULATE-3 methodology to perform nuclear design calculations including: reload design, physics input to safety analysis, physics input to fuel and clad performance, physics input to mechanical design, physics input to thermal-hydraulic analysis, input to LOCA/Non-LOCA transient analysis, CECOR coefficients, startup test predictions, core physics data books, Shutdown Margin, inputs to reactor protection system and monitoring system (COLSS/CPC) functions and setpoint and uncertainty updates, and other safety related physics parameters in support of refueling, safety analysis, and operation. These are the same functions as for the current DIT/ROCS/MC codes.

B. PURPOSE OF THE TECHNICAL SPECIFICATION

Technical Specification 5.6.5.b lists those analytical methods used to determine the core operating limits. These analytical methods are reviewed and approved by the NRC. This proposed amendment would amend this list to include an additional method for allowing APS to use CASMO-4/SIMULATE-3 in performing nuclear design calculations.

C. NEED FOR THE TECHNICAL SPECIFICATION AMENDMENT

The methods and computer codes used to analyze the nuclear design of the reactor core are described in Chapter 4 of the Palo Verde Nuclear Generating Station (PVNGS) Updated Safety Analysis Report (UFSAR). The NRC approved codes used to determine the core operating limits for TS 3.1.1, 3.1.2, 3.1.4, 3.1.7, and 3.9.1 are the

ABB/CE codes DIT and ROCS/MC, described in Technical Specification 5.6.5, Core Operating Limits Report (COLR), item b.2, "The ROCS and DIT Computer Codes for Nuclear Design," CENPD-266-P-A, April 1983.

In a continuing effort to improve reload design methods, Arizona Public Service Company (APS) has developed the CASMO-4/SIMULATE-3 reactor physics methodology. The attached topical report (Enclosure 2), "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3", September 1999, documents the validation and level of accuracy of the reactor core physics methodology to be used by APS to perform analyses for PVNGS Units 1, 2, and 3. The methodology is based on the CASMO-4/SIMULATE-3 computer program package. STUDSVIK AB and STUDSVIK of America (currently Studsvik Scandpower, Inc.) developed the CASMO-4/SIMULATE-3 computer program package. The APS methodology has been validated by benchmarkmarking CASMO-4/SIMULATE-3 predictions with measured data from PVNGS Units 1, 2, and 3 using a variety of fuel designs and operating conditions in power reactors, and Rensselaer Polytechnic Institute (RPI) and Babcock and Wilcox (B&W) critical experiments as discussed in the attached topical report (Enclosure 2).

The use of CASMO-4/SIMULATE-3 will facilitate a more efficient interface between the various reactor physics analyses. SIMULATE-3 is currently being used for Core Follow and for some portions of the Core Data Book. CASMO-4 and SIMULATE-3 are more compatible with the codes used for fuel management. SIMAN and XIMAGE are the computer programs used for fuel management at Palo Verde and are based on SIMULATE-3. Also, it is desirable to have the future capability of performing detailed space-time neutronics calculations for both design and off-design transients. SIMULATE-3K, a space-time neutronics code, can meet this objective and is built upon SIMULATE-3 using CASMO-4 cross section and kinetics parameters.

The CASMO-4/SIMULATE-3 are newer and more advanced codes. The codes provide equivalent solutions and will assist in streamlining our work process relative to our current use of DIT/ROCS/MC codes. APS currently uses CASMO-4 /SIMULATE-3 in core loading pattern development and in Core Follow operations. Thus, implementation on transient analysis allows for one consistent set of core models.

APS intends to replace the DIT/ROCS/MC methodology with CASMO-4/SIMULATE-3 while retaining the ability to use the DIT/ROCS/MC method. DIT/ROCS/MC will be maintained for the following reasons: 1) DIT/ROCS/MC method was used for determining initial safety analysis bounding parameters. 2) To maintain consistency in case the need exists to go back and rework or augment an "old" analysis. The change will not alter any methodology used in reload analysis other than replacing the DIT/ROCS/MC methodology (code package) with an equivalent methodology (code package). APS intends to use the CASMO-4/SIMULATE-3 methodology to perform all steady-state PWR core physics analyses, including:

- reload design
- physics input to safety analysis
- physics input to fuel and clad performance
- physics input to mechanical design
- physics input to thermal-hydraulic analysis
- input to LOCA/Non-LOCA transient analysis
- CECOR coefficients
- startup test predictions
- core physics data books
- Shutdown Margin
- inputs to reactor protection system and monitoring system (COLSS/CPC) functions and setpoint and uncertainty updates and
- other safety related physics parameters in support of refueling, safety analysis, and operation.

Based on the results from the APS benchmarking effort, a set of biases and uncertainties and a method for maintaining and updating these biases and uncertainties has been established. The CASMO-4/SIMULATE-3 code package produces similar and in some cases better biases and uncertainties than DIT/ROCS/MC.

APS requests that the enclosed Technical Specification amendment request be reviewed and approved by January 31, 2001, with an allowance of 45 days for implementation of the approved amendment. This review and approval schedule will support future core design reload schedules.

D. SAFETY ANALYSIS OF THE PROPOSED TECHNICAL SPECIFICATION AMENDMENT

The proposed change involves replacing the NRC approved ABB/CE codes DIT and ROCS/MC, with the Studsvik codes CASMO-4 and SIMULATE-3 while retaining the ability to use DIT/ROCS/MC. DIT/ROCS/MC is listed in Technical Specification

5.6.5.b.2, "CORE OPERATING LIMITS (COLR)" as the NRC approved code for determining the following core operating limits:

1) Specification 3.1.1, Shutdown Margin – Reactor Trip Breakers Open;
2) Specification 3.1.2, Shutdown Margin – Reactor Trip Breakers Closed;
3) Specification 3.1.4, Moderator Temperature Coefficient BOL and EOL limits;
4) Specification 3.1.7, Regulating CEA Insertion Limits; and
5) 3.9.1, Boron Concentration (Mode 6).

CASMO-4 and DIT are two-dimensional transport theory codes that provide cross-sections and perform burnup calculations on fuel assemblies or individual pin cells. ROCS/MC and SIMULATE-3 are two-group, steady-state, diffusion theory codes which perform static and depletion dependent reactor core calculations in two- or three-dimensions and full-, half-, or quarter-core symmetric geometries.

An extensive benchmark of CASMO-4/SIMULATE-3 predictions with measured data using a variety of fuel designs and operating conditions in power reactors and critical experiments, has been performed (Enclosure 2). The accuracy of CASMO-4 /SIMULATE-3 is similar to, and sometimes better than, the accuracy of DIT/ROCS/MC.

The proposed change will allow the CASMO-4/SIMULATE-3 code package to perform the same function that the DIT/ROCS/MC code package performs in calculating the physics parameters described above.

The CASMO-3/SIMULATE-3 program package has been submitted to the NRC for review and approval. Yankee Atomic Electric Company submitted Topical Reports YAEC-1363 and YAEC-1659, Southern California Edison submitted Topical Report SCE-9001, and Duke Power submitted Topical Report DPC-NE-1004. These submittals were found to be acceptable by the NRC through issued Safety Evaluation Reports (SERs) for these programs (References 1-4 in the cover letter for this submittal). Currently, the NRC has for review Northern States Power Company submittal for use of CASMO-4/SIMULATE-3 (Reference 5). Additionally, Siemens Power Corporation submitted to the NRC for review and approval a topical report for use of CASMO-4 methodology. Report "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4 /MICROBURN-B2" was submitted to the NRC on December 30, 1998 (Reference 6). This report was found to be acceptable for use only in Boiling Water Reactors (BWR), by the NRC SER (Reference 7).

Yankee Atomic Electric Company (YAEC) provided the theoretical basis and validation of the CASMO-3/SIMULATE-3 computer program package to the NRC (References 6 through 9, Enclosure 2). In these reports, YAEC provided detailed descriptions of the computer programs and a general methodology for performing reactor physics analyses. The methodology for CASMO-4 is described in Reference 2 (Enclosure 2) and verification and validation information is given in References 20 and 21 (Enclosure 2).

E. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (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. A discussion of these standards as they relate to this amendment request follows:

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

No. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Arizona Public Service Company (APS) intends to replace the DIT/ROCS/MC methodology with CASMO-4/SIMULATE-3 code package. The proposed amendment would add methodology using CASMO-4 and SIMULATE-3 codes to the list of analytical methods used to determine core operating limits contained in Technical Specification 5.6.5.b. This will allow the use of the CASMO-4 and SIMULATE-3 methodology to perform all steady-state PWR core physics analyses.

The probability of occurrence of an accident previously evaluated will not be increased by the proposed change in the particular codes used for physics calculations for nuclear design analysis. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the Updated Final Safety Analysis Report (UFSAR). These inputs do not alter the physical characteristics or modes of operation of any system, structure, or component involved in the initiation of an accident. Thus, there is no significant increase in the probability of an accident previously evaluated as a result of this change.

The consequences of an accident evaluated in the UFSAR are affected by the value of inputs to the transient safety analysis. An extensive benchmark of CASMO-4 /SIMULATE-3 predictions with measured data using a variety of fuel designs and operating conditions in power reactors and critical experiments, was performed. The accuracy of CASMO-4/SIMULATE-3 is similar to, and sometimes better than, the accuracy of DIT/ROCS/MC. Furthermore, there is always the potential for the value of the nuclear design parameters to change solely as a result of the new reload fuel core loading pattern. Regardless of the source of a change, an assessment is always made of changes to the nuclear design parameters with respect to their effects on the consequences of accidents previously evaluated in the UFSAR. Refueling is an anticipated activity which is described in the UFSAR. If increased consequences are anticipated, compensatory actions are implemented to neutralize any expected increase in consequences. These compensatory actions include, but are not limited to, crediting any existing margins in the analysis or redefining the operating envelope to avoid increased consequences. Thus, the nuclear design parameters are intermediate results and by themselves will not result in an increase in the consequence of an accident evaluated in the UFSAR.

Therefore, the replacement of the DIT/ROCS/MC codes with the CASMO-4/SIMULATE-3 code package, which will perform the same functions as the DIT/ROCS/MC codes with similar accuracy, does not significantly increase the consequences of an accident previously evaluated.

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

No. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Arizona Public Service Company (APS) intends to replace the DIT/ROCS/MC methodology with CASMO-4/SIMULATE-3 code package. The proposed amendment would add methodology using CASMO-4 and SIMULATE-3 codes to the list of analytical methods used to determine core operating limits contained in Technical Specification 5.6.5.b.

The possibility for a new or different kind of accident evaluated previously in the UFSAR will not be created by the proposed change to the particular codes used for physics calculations for nuclear design analyses. The change involves replacing the NRC approved ABB Combustion Engineering Nuclear Power (ABB/CE) DIT and ROCS/MC codes, with the Studsvik CASMO-4 and SIMULATE-3 codes. The results of nuclear design analyses are used as inputs to the analysis of accidents that are evaluated in the UFSAR. These inputs do not alter the physical characteristics or modes of operation of any system, structure or component involved in the initiation of an accident.

Therefore, the replacement of the DIT/ROCS/MC codes with the CASMO-4 /SIMULATE-3 code package, which will perform the same functions as the DIT/ROCS/MC codes with similar accuracy, does not increase the possibility of a new or different kind of accident from any accident previously evaluated.

Standard 3 -- Does the proposed change involve a significant reduction in a margin of safety?

No. The proposed change does not involve a significant reduction in a margin of safety.

The margin of safety as defined in the basis for any technical specification will not be reduced nor increased by the proposed change to the particular codes used for physics calculations for nuclear design analyses. The change involves replacing the NRC approved ABB/CE DIT and ROCS/MC codes, with the Studsvik CASMO-4 and SIMULATE-3 codes. Extensive benchmarking of the CASMO-4/SIMULATE-3 computer codes has demonstrated that the values of those parameters used in the safety analysis are not significantly changed relative to the values obtained using the DIT/ROCS/MC computer codes. For any changes in the calculated values that do occur, the application of appropriate biases and uncertainties ensures that the current margin of safety is maintained. Specifically, use of these code specific biases and uncertainties in safety evaluations continues to provide the same statistical assurance that the values of the nuclear parameters used in the safety analysis are conservative with respect to the actual values on at least a 95/95 probability/confidence basis.

Based on the responses to these three criteria, APS has concluded that the proposed amendment involves no significant hazards consideration.

F. ENVIRONMENTAL CONSIDERATION

APS has determined that the proposed amendment involves no changes in the amount or type of effluent that may be released offsite, and results in no increase in individual or cumulative occupational radiation exposure. As described above, the proposed Technical Specification amendment involves no significant hazards consideration and, as such, meets the eligibility criteria for categorical exclusion set forth in 10CFR 51.22(c)(9).

G. MARKED-UP TECHNICAL SPECIFICATION PAGE

Units 1, 2, and 3: Page 5.6-5

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

8. Letter: O.D. Parr (NRC) to F. M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for: 5.6.5.b.6.

9. Letter: K. Kniel (NRC) to A. E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 5.6.5.b.7.

10. "Fuel Rod Maximum Allowable Pressure," CEN-372-P-A, May 1990 (Methodology for Specification 3.2.1, Linear Heat Rate).

11. Letter: A. C. Thadani (NRC) to A. E. Scherer (CE), dated April 10, 1990, ("Acceptance for Reference CE Topical Report CEN-372-P"). NRC approval for 5.6.5.b.10.

Add

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 mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 PAM Report

When a report is required by Condition B or G of LCO 3.3.10, "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.

12. "Arizona Public Service Company PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3," September 1999 [Methodology for Specifications 3.1.1, Shutdown Margin - Reactor Trip Breakers Open; 3.1.2, Shutdown Margin - Reactor Trip Breakers Closed; 3.1.4, Moderator Temperature Coefficient; 3.1.7, Regulating CEA Insertion Limits and 3.9.1, Boron Concentration (Model)].

(continued)

H. RETYPED TECHNICAL SPECIFICATION PAGE

Units 1, 2, and 3: Page 5.6-5

5.6 Reporting Requirements (continued)

5.6.5 Core Operating Limits Report (COLR) (continued)

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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 mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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(continued)

ENCLOSURE 2

Arizona Public Service Company
PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3, September 1999

Proprietary Version