JAN 1 4 2004

L-PI-04-002
10 CFR 50.90

U S Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
DOCKETS 50-282 AND 50-306
LICENSE Nos. DPR-42 AND DPR-60

SUPPLEMENT TO LICENSE AMENDMENT REQUEST (LAR) DATED MARCH 25, 2003, SAFETY ANALYSES TRANSITION

By letter dated March 25, 2003, the Nuclear Management Company, LLC (NMC) submitted an LAR titled, "Safety Analyses Transition" which proposes Technical Specification (TS) changes associated with transition to Westinghouse performance of safety analyses for the Prairie Island Nuclear Generating Plant (PINGP). By letter dated November 26, 2003, the NRC Staff requested additional information for support of their review of the subject LAR. This letter provides the NMC responses to the NRC request for additional information and supplements the subject LAR. NMC submits this supplement in accordance with the provisions of 10 CFR 50.90.

Attachments 2, 3, 4, 5 and 6 provide the NMC responses to the NRC Staff request for additional information.

Attachment 3 provides 5 copies of, "Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306," dated December 9, 2003 (Proprietary).

Attachment 4 provides 5 copies of, "Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306," dated December 9, 2003 (Nonproprietary).

Also provided as Attachment 1 is Westinghouse authorization letter CAW-03-1752 with accompanying affidavit, Proprietary Information Notice, and Copyright Notice. As

Attachments 3 and 5 Contain Proprietary Information

APol 1/5

Extra Copies Forwarded to PM

Attachment 3 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations. Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse affidavit should reference CAW-03-1752 and should be addressed to J. S. Galembush, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Attachment 5 provides 5 copies of, " Safety Analysis Transition Program Licensing Report for the Prairie Island Nuclear Generating Plant," dated December 2003 (proprietary) which provides the response to RAI question 2. The enclosed report contains information which is proprietary to Westinghouse Electric Company LLC. Accordingly, Westinghouse requests that this information be withheld from public disclosure. In order not to delay this submittal of information to the Commission in connection with the review of the Licensing Report, Westinghouse will comply with the requirements of 10 CFR 2.790 to provide proprietary and non-proprietary versions together with an affidavit as soon as the proprietary information contained in the submittal have been specifically identified and the proprietary and non-proprietary versions have been prepared. NMC will submit the total required number of copies of the proprietary and non-proprietary versions of the information and the required affidavit at that time. In the meantime, NMC has provided sufficient copies for the NRC to initiate review. Representatives of the NRC office of the General Counsel have advised Westinghouse that they concur with this procedure. NMC expects to be able to fully comply with the requirements for the proprietary and non-proprietary versions of the information and an accompanying affidavit by February 15, 2004.

Attachment 7 provides the pressurizer pressure – low setpoint calculations in support of NMCs response to RAI 5. Attachment 8 provides TS pages marked up in response to RAI Question 3 and Attachment 9 provides the TS pages revised in response to RAI Question 3.

NMC requests NRC approval to implement this proposed license amendment within 90 days of NRC approval.

The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the original March 25, 2003 submittal as supplemented June 16, 2003.

Attachments 3 and 5 Contain Proprietary Information

The proposed changes in this supplement do not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the original March 25, 2003 submittal as supplemented June 16, 2003.

In accordance with 10 CFR 50.91, NMC is notifying the State of Minnesota of this LAR supplement by transmitting a copy of this letter and non-proprietary attachments to the designated State Official.

In this letter NMC has not made any new or revised any Nuclear Regulatory Commission commitments. Please address any comments or questions regarding this LAR supplement to Mr. Dale Vincent at 1-651-388-1121.

I declare under penalty of perjury that the foregoing is true and accurate. Executed on JAN 1 4 2004

Joseph M. Solymossy
Site Vice-President, Prairie Island Nuclear Generating Plant

CC Regional Administrator, USNRC, Region III
Project Manager, Prairie Island Nuclear Generating Plant, USNRC, NRR
NRC Resident Inspector – Prairie Island Nuclear Generating Plant
Glenn Wilson, State of Minnesota

Attachments:
1. Westinghouse letter CAW-03-1752 with Affidavit, Proprietary Information Notice, and Copyright Notice
2. NRC RAIs and NMC Responses
3. Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical Specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306 (Proprietary)
4. Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical Specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306 (Nonproprietary)
5. Safety Analysis Transition Program Licensing Report for the Prairie Island Nuclear Generating Plant (Proprietary)

Attachments 3 and 5 Contain Proprietary Information

6. RAI Response Tables
7 Pressurizer Pressure – Low Setpoint Calculations
8. TS pages - marked up
9. TS pages - revised

Attachments 3 and 5 Contain Proprietary Information

1717 Wakonade Drive East • Welch, Minnesota 55089-9642
Telephone: 651.388.1121

Attachment 2
L-PI-04-002
NRC RAIs and NMC Responses

NRC Request for Additional Information (RAI) in support of the Safety Analysis Transition License Amendment Request and Nuclear Management Company (NMC) Responses

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1. **Regarding the proposed use of Westinghouse safety analyses to support safe operation of the PINGP, provide a list of the titles of the topical reports that document the methodologies (including empirical correlations such as the critical heat flux correlation with its safety limit) and computer codes used in conversion to Westinghouse safety analyses for supporting the PINGP licensing application. For each listed report, (1) discuss the purposes of the methodologies and computer codes documented in the report; (2) list the associated NRC acceptance letter and any safety evaluation approving the use of the report previously issued to the PINGP; (3) address the compliance with each of the limitations imposed on use of the report; and (4) identify changes to the NRC-approved report and address acceptability of the changes.**

NMC answer to Question 1:

1. The approved topical reports documenting the methodologies and computer codes used in conversion to Westinghouse safety analyses are listed in Table RAI1-1 (Attachment 6), along with a discussion of the purposes of those methodologies and computer codes except as follows. Some safety analyses have been routinely performed by Westinghouse prior to the decision to convert to Westinghouse safety analyses and are not considered a part of the "conversion" to Westinghouse safety analyses. The topicals used for these safety analyses previously performed by Westinghouse are not included in Table RAI1-1. The topicals discussed in the March 25, 2003 LAR are not included in Table RAI1-1 since their use is addressed in the LAR. WCAP-14483 is an administrative document and is not addressed in Table RAI1-1 or the other supporting Tables.

   NMC is not transitioning to Westinghouse for containment response analyses at this time. Thus the RAI response Tables RAI1-1 through RAI1-11 provided in Attachment 6 do not include topicals related to containment response analyses. The containment response analyses are independent from the other analyses that are being transitioned to Westinghouse. They may be transitioned at a later time without impact on the analyses associated with this LAR.

2. As indicated in Tables RAI1-2 through RAI1-11, the NRC staff has approved all the listed topical reports. There have been no safety evaluations specifically approving the use of these reports issued to the PINGP because they have not been previously used for this plant. However, these codes and methods have been approved for use by similar plants (i.e., Kewaunee and Point Beach).

3. Tables RAI1-2 through RAI1-11 identify the NRC Safety Evaluation Report (SER) conditions and restrictions for each of the reports, along with the justifications for how each SER condition/restriction is satisfied in the PINGP analyses. To help ensure that all applicable SER conditions and restrictions are satisfied for each transient analysis that is performed, Westinghouse utilizes internal methodology guidance documents.

4. There have been no changes to the NRC-approved reports.

2. Regarding the PINGP specific analyses, for each event analyzed with the Westinghouse computer codes, provide the results of the transient analysis and demonstrate that the results meet the applicable acceptance criteria. For each transient analyzed, (1) describe the nodalization scheme and demonstrate its adequacy; (2) list the models and empirical correlations specified in the computer codes as "options" that are selected for the transient analysis, and demonstrate that the selected models and correlations are adequate; (3) discuss how the computer codes are used related to iteration between the codes; (4) discuss the plant initial conditions, values of plant parameters (specific to fuel design, nuclear physics and thermal-hydraulics characteristics) assumed in the transient analysis, and also justify that the values used are adequate and result in a worst case.

NMC answer to Question 2:

See "Prairie Island Units 1 and 2, Safety Analysis Transition Program, Licensing Report" included as Attachment 5 to this letter L-PI-04-002 for the response to this RAI. Section 5.1 of the Licensing Report provides details of the non-LOCA transient analyses performed by Westinghouse. Included therein for each transient are the applied computer code(s), applicable acceptance criteria, initial conditions, key assumptions, and results. The computer code nodalization schemes, options and their adequacy are discussed in the approved, code-specific topical reports (WCAPs), which are specified in the licensing report. The relationships between computer codes are also identified, as applicable, in the licensing report.

To help ensure that the assumed initial conditions and other plant parameters are conservative, and that applicable Safety Evaluation Report requirements are satisfied, Westinghouse utilizes internal methodology guidance documents known as Safety Analysis Standards (SAS) and RETRAN Safety Analysis Standards (RSAS). Each SAS/RSAS document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions, e.g., directions of conservatism for initial condition values, a detailed description of the transient model development, and a discussion of the expected transient analysis results.

The combination of the Westinghouse topical reports, SAS/RSAS documents and the PINGP specific calculations comprise the methods and results of the PINGP specific analyses. The calculations and the SAS/RSAS documents are voluminous and proprietary to Westinghouse and may be reviewed at the Westinghouse offices. NMC will provide copies of specific documents upon request.

Note, the Licensing Report discusses replacement steam generators because the Westinghouse analyses were performed to bound the replacement steam generators which NMC plans to install in the Fall of 2004.

3. Add to Technical Specification 5.6.5.b the titles of the topical reports that document applicable methodologies and computer codes used in conversion to the Westinghouse safety analysis.

NMC answer to Question 3:

The title of TS 5.6.5 is, Core Operating Limits Report (COLR), and it logically follows that the contents of this section should relate to core operating limits. The intent of this TS is to define methodologies which are used to generate core operating limits. NMC has discussed this question with the NRC to clarify the meaning and scope of this question. Based on those discussions, NMC understands that the NRC is requesting that all topical reports related to methodologies and/or computer codes, which are used to determine the core operating limits, be included in Technical Specification 5.6.5.b. Other topical reports, used in the Westinghouse safety analyses that NMC demonstrates are not used to determine the core operating limits, are not be included in TS 5.6.5.b.

Attachment 8 includes TS pages marked up to include the additional topicals identified as items 17 through 26 on pages 5.0-36 and 5.0-37 in response to this RAI. Attachment 9 includes the revised TS pages including the additional topicals.

4. On Page 4 of Exhibit A of the March 25, 2003, application, the f(ΔI) function is proposed to be removed from the OPΔT equation in Table 3.3.1-1 (page 8). This proposed change appears to be inconsistent with NUREG-1431, Revision 2, "Standard Technical Specifications -Westinghouse Plants." Table 3.3.1-1 (page 6 of 6) of the NUREG indicates that the f(ΔI) function in the OPΔT equation is a constant and can be specified in the core operating limits report (COLR). Also, page A-20 of WCAP-10216-A, Revision 1A, "Relaxation of Constant Axial Offset Control," indicates that current 17x17 plants with the constant axial offset control (CAOC) operate based on an analysis without OPΔT f(ΔI) function since OTΔT f(ΔI) function is more restrictive. If the need for an OPΔT f(ΔI) function is indicated by the relaxed axial offset control (RAOC) analysis, the OTΔT f(ΔI) function should be changed such that it should be more restrictive. Provide information to demonstrate that it is adequate to remove the f(ΔI) function from the OPΔT equation in the conversion process from the CAOC to the RAOC.

Additionally, Westinghouse has verified, via 'plant-specific analyses' for several 14X14 and 15X15 plants, that an OPΔT f(ΔI) function can be set to zero. Provide information to confirm that the 'plant-specific analyses' are applicable to PINGP with the proposed RAOC operation.

NMC answer to Question 4:

Section 3-2 of WCAP-8745-P-A discusses core power distribution effects for the overpower ΔT (OPΔT) trip, including a discussion of the historical "flyspeck" method that was originally used to determine an appropriate f(ΔI) penalty. The WCAP concludes that the newer 16x16 and 17x17 fuel assembly plant designs required no f(ΔI) penalty on OPΔT to preclude fuel centerline melting. This was true due to the use of constant axial offset control (CAOC) and the elimination of part-length (PL) control rods in the core designs. The WCAP further concludes that all operating 14x14 and 15x15 fuel assembly Westinghouse plants (at the time) had OPΔT f(ΔI) setpoints established in a more conservative manner, i.e., prior to the introduction of CAOC, and thus remained conservative.

Further information on this subject was supplied in response to a question raised during the review of WCAP-8745. On October 29, 1985, the NRC (Marv Dunenfeld) and representatives from Brookhaven National Labs (BNL) visited the Westinghouse offices in Monroeville, PA. The purpose was to discuss Westinghouse responses to a number of informal questions supplied by NRC and BNL during their review of WCAP-8745. All questions were adequately resolved, leading to the issuance of the SER on the topical. While the questions and accepted responses were not required to be formally incorporated into the

approved topical, they are documented in an internal Westinghouse letter. The relevant question and response is as follows:

> Question 3A: In Section 3-2, it is stated that no f(ΔI) function is required to preclude fuel centerline melting during overpower events at 16x16 and 17x17 fuel assembly plants. Since the ΔI bands vary from plant to plant, assuming CAOC operation, for which ΔI bands is this conclusion valid? In addition, is this conclusion valid for those plants operating with [relaxed axial offset control] RAOC?
>
> Response 3A: Note that this conclusion is probably also valid for 14x14 and 15x15 plants since PL rods are no longer permitted in those plants. At the time this conclusion was made 14x14 and 15x15 plants were already operating with OPΔT f(ΔI) functions. Since these functions were operationally non-restrictive, further investigation was not made. For 16x16 and 17x17 fueled plants, this conclusion has remained valid for all CAOC bands evaluated to date. It has also remained valid for RAOC plants. This conclusion is verified for each CAOC band widening and RAOC analysis.

Subsequently, Westinghouse has verified via plant-specific analyses for several 14x14 and 15x15 plants that an OPΔT f(ΔI) function is not required to preclude fuel centerline melting, and thus can be set to zero or eliminated. This is consistent with the use of either CAOC or RAOC and no PL rods in the core, and is explicitly confirmed via analysis for each plant. Westinghouse developed OPΔT setpoints as shown in Section 5 of Attachment 5, specifically in Table 5.1-4. The analyses have been completed, as discussed in Attachment 5, without an f(ΔI) penalty on OPΔT. These analyses confirmed that the OPΔT trip function will preclude fuel centerline melting without the need for an f(ΔI) penalty function.

Most plant Technical Specifications show separate f(ΔI) functions for OTΔT and OPΔT, with the f(ΔI) = 0 for OPΔT. However, older plants like Prairie Island have a single f(ΔI) function generator that is shared by both the OTΔT and OPΔT trip functions. Since an f(ΔI) function is required for OTΔT, it is necessary to physically remove this input in order to eliminate it (i.e., set it to zero) for OPΔT. Thus, it is proposed to completely remove the f(ΔI) term from Table 3.3.1-1, Note 2, in the Prairie Island Technical Specifications. Based on Westinghouse experience as noted above, no future need for this function is anticipated for Prairie Island.

5. **On Page 5 of Exhibit A of the March 25, 2003, application, the allowable value of the pressurizer pressure-low reactor trip is changed from 1760 psig to 1845 psig. Provide the rationale for the change. Specify the actual plant pressurizer pressure-low reactor trip setting and provide analyses**

which verify that with the increased allowable value, from 1760 psig to 1845 psig, the actual plant setting, with consideration of the instrumentation uncertainties, remains unchanged. Also, identify the analytical value used in the non-loss-of-coolant-accident analysis with the Westinghouse methodologies and computer codes and provide the analysis which shows that this analytical value is lower than the proposed increased allowable value of 1845 psig.

NMC answer to Question 5:

The proposed change to this value is in the conservative direction. As described and illustrated in Appendix B of WCAP-8745-P-A, the low and high pressurizer pressure reactor trips limit the range of conditions over which the OTΔT and OPΔT trips must provide protection. The more limited this range is, the less restrictive the setpoint constants need to be, which provides increased operational flexibility while still ensuring that safety limits are met. The current Allowable Value for pressurizer pressure – low is much lower than the actual plant setting. To credit and thus derive the benefit from this higher actual plant setting in the ΔT trip setpoints it is necessary to increase the safety analysis setpoint, which makes it necessary to formally increase the Allowable Value for pressurizer pressure – low in the Technical Specifications.

No change is planned for the actual plant setpoint. The following table summarizes the various setpoint values.

Pressurizer Pressure - Low Reactor Trip Limits	Current	Revised
Actual Plant Setting	1900 psig	1900 psig (no change)
Technical Specification Allowable Value	1760 psig	1845 psig
Safety Analysis Limit	1700 psig	1835 psig

The difference between the actual plant setting and the TS allowable value provides sufficient margin for calibration tolerances and measurable instrument uncertainties such as instrument accuracy, setting drift and so on. The calculations which support the nominal plant setting are provided in Attachment 7. The Westinghouse safety analyses, as shown in Attachment 5, Table 5.1-4, use 1835 psig (1850 psia) which is below the proposed TS Allowable Value of 1845 psig.

6. On Page 5 of Exhibit A of the March 25, 2003, application, it is indicated that the sensitivity factors, operating parameters, and VIPRE model used in the safety analysis do not differ significantly from those used in WCAP-11397. Provide the results of the analysis that determines the design limit DNBR using the revised thermal design procedure. The information should include a table similar to Table 3-1 of WCAP-11397, including values of the DNBR design limit, operating parameters, measurement uncertainties and sensitivity factors.

NMC answer to Question 6:

The response to this RAI contains information that has been identified by Westinghouse as proprietary information. The proprietary version of the response is included as Attachment 3 and the non-proprietary response is included as Attachment 4 to this letter, L-PI-04-002.

ATTACHMENT 4

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-04-002, Supplement to
License Amendment Request dated March 25, 2003

Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306 (Nonproprietary).

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION NUMBER 6 REGARDING THE REVIEW OF THE USE OF WESTINGHOUSE SAFETY ANALYSES AND ASSOCIATED TECHNICAL SPECIFICATION CHANGES PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNITS 1 AND 2 DOCKET NOS. 50-282 AND 50-306

DECEMBER 9, 2003

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

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6. On Page 5 of EXHIBIT A of the March 25, 2003 application, it is indicated that the sensitivity factors, operating parameters, and VIPRE model used in the safety analysis do not differ significantly from those used in WCAP-11397. Provide the results of the analysis that determines the design limit departure from nucleate (DNBR) using the revised thermal design procedure. The information should include a table similar to Table 3-1 of WCAP-11397, including values of the DNBR design limit, operating parameters, measurement uncertainties, and sensitivity factors.

The Design Limit DNBR was calculated according to the approved RTDP methodology (WCAP-11397-P-A), where the DNB correlation statistics are convoluted with plant instrumentation uncertainties to calculate the RTDP 95/95 DNBR limit.

RTDP uncertainties

The RTDP instrumentation standard deviations and WRB-1 DNB correlation statistics are presented in Table 1:

a,b,c

[ ]

Pressure, temperature, power and flow uncertainties were chosen such as to bound the uncertainties that were transmitted by NMC to Westinghouse for the Prairie Island Units.
WRB-1 statistics (mean, standard deviation and Owen factor) with VIPRE-W are from WCAP 14565-P-A.

Operating Parameters

RTDP operating parameters were chosen as follows:

Flow = 182500 gpm (MMF)
PWRINP = 1650 MWt
Tin = 528.7 deg-F (based on MMF)
FdH = 1.70 (design value – 4% uncertainties, the 4% uncertainties are accounted in Table 1)

However, different plant conditions (high power-high pressure, high power-low pressure, low power-high pressure, low power-low pressure, Loss of Flow, Static Rod Misalignment, Dropped rod and Axial Offset

Limits) were investigated in order to calculate a bounding set of DNBR and quality sensitivities (si in Table 2). [The Static Rod Misalignment conditions were found to be the most sensitive and the corresponding DNBR sensitivities were used along with the instrumentation uncertainties and DNBR correlation statistics to calculate the Design Limit DNBR (Table 2)][a,c]. Note that the DNBR sensitivities are similar to the one calculated in WCAP-11397.

[ ] a.b.c

ATTACHMENT 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-04-002, Supplement to
License Amendment Request dated March 25, 2003

Westinghouse letter CAW-03-1752
Affidavit
Proprietary Information Notice
Copyright Notice

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Our ref: CAW-03-1752

December 12, 2003

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical Specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306, dated December 9, 2003 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-03-1752 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Nuclear Management Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-03-1752, and should be addressed to J. S. Galembush, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: D. Holland
B. Benney
E. Peyton

bcc: J. S. Galembush (ECE 4-7A) 1L
R. Bastien, 1L, 1A (Nivelles, Belgium)
C. Brinkman, 1L, 1A (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)

CAW-03-1752

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. S. Galembush, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. S. Galembush, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 12th day
of December, 2003

Notary Public

SHARON L. FIORI
NOTARY PUBLIC
COMMONWEALTH OF PENNSYLVANIA

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Response to Request for Additional Information Number 6 Regarding the Review of the Use of Westinghouse Safety Analyses and Associated Technical Specification Changes, Prairie Island Nuclear Generating Plant, Units 1 and 2, Docket Nos. 50-282 and 50-306," dated December 9, 2003 (Proprietary), being transmitted by the Nuclear Management Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Prairie Island Units 1 & 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of the use of thermal hydraulic analyses to support continued safe operation.

This information is part of that which will enable Westinghouse to:

(a) Perform thermal hydraulic analyses.

(b) Assist the customer to obtain NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of continued safe operation based on its thermal hydraulic analyses.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

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ATTACHMENT 6
PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Letter L-PI-04-002, Supplement to
License Amendment Request dated March 25, 2003

RAI Response Tables

RAI Response Tables

Table RAI1-1
Topical Reports Used In Safety Analyses for Prairie Island

Report Number	Title	Purpose
WCAP-14882-P-A	RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses	The RETRAN code is being used in the analysis of the following events: ▪ Uncontrolled RCCA Withdrawal at Power ▪ Feedwater System Malfunction ▪ Loss of Reactor Coolant Flow ▪ Locked Rotor ▪ Loss of External Electrical Load ▪ Loss of Normal Feedwater ▪ Loss of AC Power to the Plant Auxiliaries ▪ Steam Line Break
WCAP-7979-P-A	TWINKLE – A Multidimensional Neutron Kinetics Computer Code	The TWINKLE code is being used in the analysis of the following events: ▪ Uncontrolled RCCA Withdrawal from a Subcritical Condition ▪ Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)
WCAP-7908-A	FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod	The FACTRAN code is being used in the analysis of the following events: ▪ Uncontrolled RCCA Withdrawal from a Subcritical Condition ▪ Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection)

Table RAI1-1 (Continued)

Report Number	Title	Purpose
WCAP-14565-P-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis	The VIPRE code is being used in the analysis of the following events: ▪ Uncontrolled RCCA Withdrawal from a Subcritical Condition ▪ RCCA Misalignment ▪ Feedwater System Malfunctions ▪ Loss of Reactor Coolant Flow ▪ Locked Rotor ▪ Steam Line Break
WCAP-10965-P-A	ANC: A Westinghouse Advanced Nodal Computer Code	The ANC code is used on a cyclic basis to generate best estimate core models, and using those models, to compare predicted parameters to limits associated with many different analyses. These comparisons include, but are not limited to: • Core reactivity response • Power distributions, in the form of FdH and FQ • Axial offset predictions • Integral and differential rod worth predictions under a variety of conditions • Reactivity coefficient predictions versus power, temperature, density, and boron under a variety of conditions
WCAP-11596-P-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores	The PHOENIX-P code is used on a cyclic basis to generate nuclear cross section input for use in the ANC code.

Table RAI1-1 (Continued)

Report Number	Title	Purpose
WCAP-11394-P-A	Methodology for the Analysis of the Dropped Rod Event	This methodology is being used in the analysis of the Dropped Rod event.
WCAP-7588 Rev. 1 –A	An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods	This methodology is being used in the analysis of the Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) event.
WCAP-12910 Rev. 1-A	Pressurizer Safety Valve Set Pressure Shift	This methodology is being used in assessing and applying the Pressurizer Safety Valve response to the plant safety analyses, specifically for transients for which it is conservative to model minimum pressurizer pressure relief.
WCAP-7907-P-A	LOFTRAN Code Description	The LOFTRAN code is being used in the analysis of the rod drop transient.

Table RAI1-2: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN

Computer Code:	RETRAN
Licensing Topical Report:	WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
Date of NRC Acceptance:	February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"The transients and accidents that Westinghouse proposes to analyze with RETRAN are listed in this SER (Table 1) and the NRC staff review of RETRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification."***

Justification

The transients listed in Table 1 of the SER are:

Feedwater system malfunctions,
Excessive increase in steam flow,
Inadvertent opening of a steam generator relief or safety valve,
Steam line break,
Loss of external load/turbine trip,
Loss of offsite power,
Loss of normal feedwater flow,
Feedwater line rupture,
Loss of forced reactor coolant flow,
Locked reactor coolant pump rotor/sheared shaft,
Control rod cluster withdrawal at power,
Dropped control rod cluster/dropped control bank,
Inadvertent increase in coolant inventory,
Inadvertent opening of a pressurizer relief or safety valve,
Steam generator tube rupture.

The transients analyzed for Prairie Island using RETRAN are:

Uncontrolled RCCA withdrawal at power (USAR 14.4.2),
Excessive heat removal due to feedwater system malfunction (USAR 14.4.6),
Loss of reactor coolant flow (USAR 14.4.8),
Locked pump rotor (USAR 14.4.8),
Loss of external electrical load (USAR 14.4.9),
Loss of normal feedwater (USAR 14.4.10),
Loss of all AC power to the station auxiliaries (USAR 14.41.11),
Rupture of a steam pipe (USAR 14.5.5).

As each transient analyzed for Prairie Island using RETRAN matches one of the transients listed in Table 1 of the SER, additional justification is not required.

Table RAI1-2: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - RETRAN

Computer Code:	RETRAN
Licensing Topical Report:	WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.
Date of NRC Acceptance:	February 11, 1999 (SER from F. Akstulewicz (NRC) to H. Sepp (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

2. ***"WCAP-14882 describes modeling of Westinghouse designed 4-, 3, and 2-loop plants of the type that are currently operating. Use of the code to analyze other designs, including the Westinghouse AP600, will require additional justification."***

Justification

The Prairie Island Nuclear Generating Plant is a 2-loop Westinghouse-designed plant that was "currently operating" at the time the SER was written (February 11, 1999). Therefore, additional justification is not required.

3. ***"Conservative safety analyses using RETRAN are dependent on the selection of conservative input. Acceptable methodology for developing plant-specific input is discussed in WCAP-14882 and in Reference 14 [WCAP-9272-P-A]. Licensing applications using RETRAN should include the source of and justification for the input data used in the analysis."***

Justification

The input data used in the RETRAN analyses performed by Westinghouse came from both NMC and Westinghouse sources. Assurance that the RETRAN input data is conservative for Prairie Island is provided via Westinghouse's use of transient-specific analysis guidance documents. Each analysis guidance document provides a description of the subject transient, a discussion of the plant protection systems that are expected to function, a list of the applicable event acceptance criteria, a list of the analysis input assumptions (e.g., directions of conservatism for initial condition values), a detailed description of the transient model development method, and a discussion of the expected transient analysis results. Based on the analysis guidance documents, conservative, plant-specific input values were requested and collected from the responsible NMC and Westinghouse sources. Consistent with the Westinghouse Reload Evaluation Methodology described in WCAP-9272-P-A, the safety analysis input values used in the Prairie Island analyses were selected to conservatively bound the values expected in subsequent operating cycles.

Table RAI1-3: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - TWINKLE

Computer Code:	TWINKLE
Licensing Topical Report:	WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975.
Date of NRC Acceptance:	July 29, 1974 (SER from D. B. Vassallo (U.S. Atomic Energy Commission) to R. Salvatori (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

There are no conditions, restrictions, or limitations cited in the TWINKLE SER.

Justification

As the TWINKLE SER does not cite any conditions, restrictions, or limitations, additional justification is not required.

Table RAI1-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN

Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"The fuel volume-averaged temperature or surface temperature can be chosen at a desired value which includes conservatisms reviewed and approved by the NRC."***

   Justification

   The FACTRAN code was used in the analyses of the following transients for Prairie Island: Uncontrolled RCCA Withdrawal from a Subcritical Condition (USAR 14.4.1) and Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection, USAR 14.5.6). Initial fuel temperatures used as FACTRAN input in the RCCA Ejection analyses were calculated using the NRC-approved PAD 3.4 computer code (see WCAP-10851-P-A). As indicated in WCAP-10851-P-A, the NRC has approved the method of determining uncertainties for PAD 3.4 fuel temperatures.

2. ***"Table 2 presents the guidelines used to select initial temperatures."***

   Justification

   In summary, Table 2 of the SER specifies that the initial fuel temperatures assumed in the FACTRAN analyses of the following transients should be "High" and include uncertainties: Loss of Flow, Locked Rotor, and Rod Ejection. As discussed above, fuel temperatures were used as input to the FACTRAN code in the RCCA Ejection analyses for Prairie Island. The assumed fuel temperatures, which were calculated using the PAD 3.4 computer code (see WCAP-10851-P-A), include uncertainties and are conservatively high. FACTRAN was not used in the Loss of Flow and Locked Rotor analyses.

3. ***"The gap heat transfer coefficient may be held at the initial constant value or can be varied as a function of time as specified in the input."***

   Justification

   The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2. For the RCCA Withdrawal from a Subcritical Condition transient, the gap heat transfer coefficient is kept at a conservative constant value throughout the transient; a high constant value is assumed to maximize the peak heat flux (for DNB concerns) and a low constant value is assumed to maximize fuel temperatures. For the RCCA Ejection transient, the initial gap heat transfer coefficient is based on the predicted initial fuel surface temperature, and is ramped rapidly to a very high value at the beginning of the transient to simulate clad collapse onto the fuel pellet.

Table RAI1-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN

Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

4. ***"...the Bishop-Sandberg-Tong correlation is sufficiently conservative and can be used in the FACTRAN code. It should be cautioned that since these correlations are applicable for local conditions only, it is necessary to use input to the FACTRAN code which reflects the local conditions. If the input values reflecting average conditions are used, there must be sufficient conservatism in the input values to make the overall method conservative."***

Justification

Local conditions related to temperature, heat flux, peaking factors and channel information were input to FACTRAN for each transient analyzed for Prairie Island (RCCA Withdrawal from a Subcritical Condition (USAR 14.4.1) and RCCA Ejection (USAR 14.5.6)). Therefore, additional justification is not required.

5. ***"The fuel rod is divided into a number of concentric rings. The maximum number of rings used to represent the fuel is 10. Based on our audit calculations we require that the minimum of 6 should be used in the analyses."***

Justification

At least 6 concentric rings were assumed in FACTRAN for each transient analyzed for Prairie Island (RCCA Withdrawal from a Subcritical Condition (USAR 14.4.1) and RCCA Ejection (USAR 14.5.6)).

6. ***"Although time-<u>Independent</u> mechanical behavior (e.g., thermal expansion, elastic deformation) of the cladding are considered in FACTRAN, time-<u>dependent</u> mechanical behavior (e.g., plastic deformation) is not considered in the code. ...for those events in which the FACTRAN code is applied (see Table 1), significant time-dependent deformation of the cladding is not expected to occur due to the short duration of these events or low cladding temperatures involved (where DNBR Limits apply), or the gap heat transfer coefficient is adjusted to a high value to simulate clad collapse onto the fuel pellet."***

Justification

The two transients that were analyzed with FACTRAN for Prairie Island (RCCA Withdrawal from a Subcritical Condition (USAR 14.4.1) and RCCA Ejection (14.5.6)) are included in the list of transients provided in Table 1 of the SER; each of these transients is of short duration. For the RCCA Withdrawal from a Subcritical Condition transient, relatively low cladding temperatures are involved, and the gap heat transfer coefficient is kept constant throughout the transient. For the RCCA Ejection transient, a high gap heat transfer coefficient is applied to simulate clad collapse onto the fuel pellet. The gap heat transfer coefficients applied in the FACTRAN analyses are consistent with SER Table 2.

Table RAI1-4: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - FACTRAN

Computer Code:	FACTRAN
Licensing Topical Report:	WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.
Date of NRC Acceptance:	September 30, 1986 (SER from C. E. Rossi (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

7. ***"The one group diffusion theory model in the FACTRAN code slightly overestimates at beginning of life (BOL) and underestimates at end of life (EOL) the magnitude of flux depression in the fuel when compared to the LASER code predictions for the same fuel enrichment. The LASER code uses transport theory. There is a difference of about 3 percent in the flux depression calculated using these two codes. When [T(centerline) – T(Surface)] is on the order of 3000°F, which can occur at the hot spot, the difference between the two codes will give an error of 100°F. When the fuel surface temperature is fixed, this will result in a 100°F lower prediction of the centerline temperature in FACTRAN. We have indicated this apparent nonconservatism to Westinghouse. In the letter NS-TMA-2026, dated January 12, 1979, Westinghouse proposed to incorporate the LASER-calculated power distribution shapes in FACTRAN to eliminate this non-conservatism. We find the use of the LASER-calculated power distribution in the FACTRAN code acceptable."***

Justification

The condition of concern (T(centerline – T(surface) on the order of 3000°F) is expected for transients that reach, or come close to, the fuel melt temperature. As this applies only to the RCCA ejection transient, the LASER-calculated power distributions were used in the FACTRAN analysis of the RCCA ejection transient for Prairie Island.

Table RAI1-5: Approval Status & SER Requirements for Core Analysis Codes - VIPRE

Computer Code:	VIPRE
Licensing Topical Report:	WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999.
Date of NRC Acceptance:	January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"Selection of the appropriate CHF correlation, DNBR limit, engineered hot channel factors for enthalpy rise and other fuel-dependent parameters for a specific plant application should be justified with each submittal."***

   Justification

   The WRB-1 correlation with a 95/95 correlation limit of 1.17 was used in the DNB analyses for the Prairie Island OFA fuel. The use of the WRB-1 DNB correlation for the 14x14 OFA fuel was approved in June 29, 1984 (Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), "SER on the Applicability or WRB-1 to Westinghouse 14x14 and 15x15 OFA").

   The use of the plant specific hot channel factors and other fuel dependent parameters in the DNB analysis for the Prairie Island OFA fuel were justified using the same methodologies as for previously approved safety evaluations of other Westinghouse two-loop plants using the same fuel design.

2. ***"Reactor core boundary conditions determined using other computer codes are generally input into VIPRE for reactor transient analyses. These inputs include core inlet coolant flow and enthalpy, core average power, power shape and nuclear peaking factors. These inputs should be justified as conservative for each use of VIPRE."***

   Justification

   The core boundary conditions for the VIPRE calculations for the OFA fuel are all generated from NRC-approved codes and analysis methodologies. Conservative reactor core boundary conditions were justified for use as input to VIPRE. Continued applicability of the input assumptions is verified on a cycle-by-cycle basis using the Westinghouse reload methodology WCAP-9272-P-A.

3. ***"The NRC Staff's generic SER for VIPRE set requirements for use of new CHF correlations with VIPRE. Westinghouse has met these requirements for using WRB-1, WRB-2 and WRB-2M correlations. The DNBR limit for WRB-1 and WRB-2 is 1.17. The WRB-2M correlation has a DNBR limit of 1.14. Use of other CHF correlations not currently included in VIPRE will require additional justification."***

   Justification

   As discussed in response to Condition 1, the WRB-1 correlation with a limit of 1.17 was used for the DNB analyses of OFA fuel in Prairie Island. For conditions where WRB-1 is not applicable, the W-3 DNB correlation was used with a limit of 1.30 (1.45 if 500 < P < 1000).

Table RAI1-5: Approval Status & SER Requirements for Core Analysis Codes - VIPRE

Computer Code:	VIPRE
Licensing Topical Report:	WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis" October 1999.
Date of NRC Acceptance:	January 19, 1999 (SER from T. H. Essig (NRC) to H. Sepp (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

4. ***"Westinghouse proposes to use the VIPRE code to evaluate fuel performance following postulated design-basis accidents, including beyond-CHF heat transfer conditions. These evaluations are necessary to evaluate the extent of core damage and to ensure that the core maintains a coolable geometry in the evaluation of certain accident scenarios. The NRC Staff's generic review of VIPRE did not extent to post CHF calculations. VIPRE does not model the time-dependent physical changes that may occur within the fuel rods at elevated temperatures. Westinghouse proposes to use conservative input in order to account for these effects. The NRC Staff requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to ensure that conservative results are obtained."***

Justification

For application to Prairie Island safety analysis, the usage of VIPRE in the post-CHF region is limited to the peak clad temperature calculation for the locked rotor transient. The calculation demonstrated that the peak clad temperature in the reactor core is well below the allowable limit to prevent clad embrittlement. VIPRE modeling of the fuel rod is consistent with the model described in WCAP-14565-P-A and included the following conservative assumptions:

- DNB was assumed to occur at the beginning of the transient
- Film boiling was calculated using the Bishop-Sandberg-Tong correlation
- The Baker-Just correlation accounted for heat generation in fuel cladding due to zirconium-water reaction.

Conservative results were further ensured with the following input:

- Fuel rod input based on the maximum fuel temperature at the given power
- The hot spot power factor was equal to or greater than the design linear heat rate
- Uncertainties were applied to the initial operating conditions in the limiting direction

Table RAI1-6: Approval Status & SER Requirements for Core Analysis Codes - ANC

Computer Code:	ANC
Licensing Topical Report:	WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code" September 1986.
Date of NRC Acceptance:	June 23, 1986 (SER from C. Berlinger (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

Although there are no conditions, restrictions, or limitations explicitly cited in the ANC SER, the SER does conclude that "the ANC code provides an accurate calculation of core reactivity, reactivity coefficients critical boron, rod worths and core power distribution for use in design and safety analyses."

Justification

In support of the Prairie Island analysis, the ANC code was used to calculate power distributions for normal (design) and off-normal (safety analysis) conditions, and was also used for reactivity calculations. As these code applications are consistent with those listed in the SER, additional justification is not required.

Table RAI1-7: Approval Status & SER Requirements for Core Analysis Codes – PHOENIX-P

Computer Code:	PHOENIX-P
Licensing Topical Report:	WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores" June 1988.
Date of NRC Acceptance:	May 17, 1988 (SER from A. C. Thadani (NRC) to R. W. Johanson (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

Although there are no conditions, restrictions, or limitations explicitly cited in the PHOENIX-P/ANC SER, the SER does conclude that "The PHOENIX-P lattice physics methods and the PHOENIX-P/ANC nodal analysis systems described in this report are thus acceptable for use in PWR design analysis.".

Justification

As Prairie Island is a PWR plant, it is acceptable to use the PHOENIX-P and PHOENIX-P/ANC code system, and additional justification is not required.

Table RAI1-8: Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – Dropped Rod

Transient:	RCCA Misalignment (Dropped Rod)
Licensing Topical Report:	WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.
Date of NRC Acceptance:	October 23, 1989 (SER from A. C. Thadani (NRC) to R. A. Newton (WOG))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"The Westinghouse analysis, results and comparisons are reactor and cycle specific. No credit is taken for any direct reactor trip due to dropped RCCA(s). Also, the analysis assumes no automatic power reduction features are actuated by the dropped RCCA(s). A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results."***

Justification

For the reference cycle assumed in the Prairie Island safety analysis transition program, it will be affirmed that the methodology described in WCAP-11394-P-A was performed and the required comparisons made with acceptable results (DNB limits are not exceeded).

Table RAI1-9: Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – RCCA Ejection

Transient:	RCCA Ejection
Licensing Topical Report:	WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.
Date of NRC Acceptance:	August 28, 1973 (SER from D. B. Vassallo (AEC) to R. Salvatori (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"The staff position, as well as that of the reactor vendors over the last several years, has been to limit the average fuel pellet enthalpy at the hot spot following a rod ejection accident to 280 cal/gm. This was based primarily on the results of the SPERT tests which showed that, in general, fuel failure consequences for UO_2 have been insignificant below 300 cal/gm for both irradiated and unirradiated fuel rods as far as rapid fragmentation and dispersal of fuel and cladding into the coolant are concerned. In this report, Westinghouse has decreased their limiting fuel failure criterion from 280 cal/gm (somewhat less than the threshold of significant conversion of the fuel thermal energy to mechanical energy) to 225 cal/gm for unirradiated rods and 200 cal/gm for irradiated rods. Since this is a conservative revision on the side of safety, the staff concludes that it is an acceptable fuel failure criterion."***

   Justification

   The maximum fuel pellet enthalpy at the hot spot calculated for each Prairie Island specific RCCA Ejection case is less than 200 cal/gm. These results satisfy the fuel failure criterion accepted by the staff.

2. ***"Westinghouse proposes a clad temperature limitation of 2700°F as the temperature above which clad embrittlement may be expected. Although this is several hundred degrees above the maximum clad temperature limitation imposed in the AEC ECCS Interim Acceptance Criteria, this is felt to be adequate in view of the relatively short time at temperature and the highly localized effect of a reactivity transient."***

   Justification

   As discussed in Westinghouse letter NS-NRC-89-3466 written to the NRC (W. J. Johnson to R. C. Jones, dated October 23, 1989), the 2700°F clad temperature limit was historically applied by Westinghouse to demonstrate that the core remains in a coolable geometry during an RCCA ejection transient. This limit was never used to demonstrate compliance with fuel failure limits and is no longer used to demonstrate core coolability. The RCCA ejection acceptance criteria applied by Westinghouse to demonstrate long term core coolability and compliance with applicable offsite dose requirements are the following.

   - Maximum average fuel pellet enthalpy at the hot spot must remain below 200 cal/gm.
   - Peak RCS pressure must remain below that which would cause the stresses in the RCS to exceed the Faulted Condition stress limits.
   - Maximum fuel melting must be limited to the innermost 10% of the fuel pellet at the hot spot, independent of the above pellet enthalpy limit.

Table RAI1-10: Approval Status & SER Requirements for Non-LOCA Transient Analysis Methods – Pressurizer Safety Valve Response

Transient:	Rod Withdrawal at Power (Peak RCS Pressure Case), Locked Rotor (Peak RCS Pressure Case), Loss of Load (Peak RCS Pressure Case)
Licensing Topical Report:	WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift," May 1993.
Date of NRC Acceptance:	February 19, 1993 (SER from J. E. Richardson (NRC) to T. E. Herrmann (Westinghouse Owners Group))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"The staff has also determined that the proposed methodology for applying the PSV performance to the plant accident analyses is acceptable with the following provision. As stated above, for events where there is additional RCS pressure increase after the PSVs have fully opened, this additional pressure should be considered in determining the maximum tolerable purge time."***

   Justification

   A minimum tolerable purge time is not determined for the Prairie Island analyses. Rather, a conservative plant specific loop seal purge time was calculated for the Prairie Island PSV configuration based on the methodology described in Section 9.3 of WCAP-12910, and directly applied in the relevant safety analyses.

2. ***"The NRC staff approval of the topical report is limited to the PSV set pressure shift assessment and methodology for applying the PSV response to the plant safety analyses."***

   Justification

   The use of the topical report in the Prairie Island safety analyses has been limited to the PSV set pressure shift assessment and methodology for applying the PSV response to the plant safety analyses.

Table RAI1-11: Approval Status & SER Requirements for Non-LOCA Transient Analysis Codes - LOFTRAN

Computer Code:	LOFTRAN
Licensing Topical Report:	WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.
Date of NRC Acceptance:	July 29, 1983 (SER from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse))

Safety Evaluation Report (SER) Conditions & Justification for the Prairie Island Nuclear Generating Plant

1. ***"LOFTRAN is used to simulate plant response to many of the postulated events reported in Chapter 15 of PSARs and FSARs, to simulate anticipated transients without scram, for equipment sizing studies, and to define mass/energy releases for containment pressure analysis. The Chapter 15 events analyzed with LOFTRAN are:***

   - ***Feedwater System Malfunction***
   - ***Excessive Increase in Steam Flow***
   - ***Inadvertent Opening of a Steam Generator Relief or Safety Valve***
   - ***Steamline Break***
   - ***Loss of External Load***
   - ***Loss of Offsite Power***
   - ***Loss of Normal Feedwater***
   - ***Feedwater Line Rupture***
   - ***Loss of Forced Reactor Coolant Flow***
   - ***Locked Pump Rotor***
   - ***Rod Withdrawal at Power***
   - ***Rod Drop***
   - ***Startup of an Inactive Pump***
   - ***Inadvertent ECCS Actuation***
   - ***Inadvertent Opening of a Pressurizer Relief or Safety Valve***

   This review is limited to the use of LOFTRAN for the licensee safety analyses of the Chapter 15 events listed above, and for a steam generator tube rupture..."

   Justification

   The LOFTRAN code was only used in the analysis of the Rod Drop transient (USAR 14.4.3) for Prairie Island. As this transient matches one of the transients listed in the SER, additional justification is not required.