NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

SEP 1 2004

L-PI-04-017
10 CFR 50.90

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

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

License Amendment Request (LAR)
Request for Use of GOTHIC 7 In Containment Response Analyses

Pursuant to 10 CFR 50.90, the Nuclear Management Company, LLC (NMC) hereby requests an amendment to allow use of the code for Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7) to model Prairie Island Nuclear Generating Plant (PINGP) containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents. The current PINGP containment response analyses are performed utilizing CONTEMPT.

The GOTHIC 7 containment evaluation model proposed for the PINGP analyses was constructed based on the recently accepted Kewaunee Nuclear Power Plant (KNPP) model associated with the KNPP License Amendment dated September 29, 2003. As discussed in Exhibits C and D, the KNPP containment evaluation model was initially developed with GOTHIC version 7.0patch2 and submitted for Nuclear Regulatory Commission (NRC) review. The NRC conditions for acceptance, primarily the change from the mist diffusion layer model to the diffusion layer model correlation, along with improvements and error corrections, were incorporated into GOTHIC version 7.1patch1 and the subsequent analyses for the KNPP were performed with that version. In complying with the conditions discussed in the NRC Safety Evaluation Report for the KNPP License Amendment, the following have not been used in PINGP analyses:

- The “height-effect” scaling factor applied to the heat and mass transfer analogy.
- The Gido-Koestel correlation.
- The inclusion of mist in the mist diffusion layer model.

Exhibit A contains the licensee's evaluation of this LAR. This LAR proposes changes to the PINGP licensing basis and does not include any material changes to the Facility Operating License, Technical Specifications (TS) or TS Bases. As discussed below,

Exhibit C Contains Proprietary Information

APOI

Exhibit B provides an affidavit from Westinghouse Electric Company for withholding Exhibit C from public disclosure. Exhibit C is the proprietary version of WCAP-16219-P, which provides a description of GOTHIC 7, the methods and restrictions for its application to PINGP, sample calculation results and comparison to current modeling results. Exhibit D is the non-proprietary version of WCAP-16219.

Once approved, the GOTHIC 7 model could replace the current CONTEMPT containment evaluation models and be used to perform the containment pressure and temperature analyses for LOCA and MSLB. This would support NMC's transition option from internal analyses using CONTEMPT to an external analyses vendor (Westinghouse), which supports GOTHIC 7.

NMC requests that the NRC review and approve the proposed license amendment by September 1, 2005. If you have any questions or require additional information, please contact Mr. Robert J Alexander at 651-388-1121.

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

Summary of Commitments

The letter contains no new commitments and no revisions to existing commitments.

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

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant Units 1 and 2
Nuclear Management Company, LLC

cc: Regional Administrator, USNRC, Region III
PINGP Senior Resident Inspector, USNRC
NRR Project Manager, USNRC
Glenn Wilson, State of Minnesota

Exhibits:
A. Evaluation of the Proposed Changes
B. Notarized Affidavit from Westinghouse dated April 29, 2004
C. WCAP-16219-P "Development and Qualification of a GOTHIC Containment Evaluation Model for the Prairie Island Nuclear Generating Plants"; March 2004 (Proprietary Version)
D. WCAP-16219-NP "Development and Qualification of a GOTHIC Containment Evaluation Model for the Prairie Island Nuclear Generating Plants"; March 2004 (Non-proprietary Version)

EXHIBIT A
LICENSEE EVALUATION

REQUEST FOR USE OF GOTHIC 7 IN CONTAINMENT RESPONSE ANALYSES

1.0 DESCRIPTION

This license amendment request (LAR) is a request to amend Operating Licenses DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant (PINGP) Units 1 and 2.

The proposed change would revise the Operating Licenses' licensing basis to allow use of the code for Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7) to model PINGP containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents. The Nuclear Management Company (NMC) is making this request to support a transition option from internal analyses using CONTEMPT to an external analyses vendor (Westinghouse), which supports GOTHIC 7.

2.0 PROPOSED CHANGE

PINGP containment response for LOCAs was performed in accordance with the methodology described in the PINGP USAR, Appendix K using the CONTEMPT analysis code. The current PINGP containment response for MSLBs was performed utilizing CONTEMPT as described in PINGP USAR Section 14.5.5. This request is to allow use of the code for Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7), as an option to model PINGP containment response for both LOCA and MSLB accidents.

This LAR proposes changes to the PINGP licensing basis and does not include any material changes to the Facility Operating Licenses, Technical Specifications (TS) or TS Bases. In summary, this change will revise the PINGP licensing basis to include the GOTHIC 7 analysis methodology for modeling containment response for LOCA and MSLB accidents.

3.0 BACKGROUND

PINGP is a two-unit plant located on the west bank of the Mississippi River approximately 6 miles northwest of the city of Red Wing, Minnesota. The facility is owned by Northern States Power Company (NSP) and operated by the NMC. Each unit at PINGP employs a two-loop pressurized water reactor designed and supplied by Westinghouse Electric Corporation. The initial PINGP application for a Construction Permit and Operating License was submitted to the Atomic Energy Commission (AEC) in April 1967. The Final Safety Analysis Report (FSAR) was submitted for application of an Operating License in January 1971. Prairie Island Unit 1 began commercial operation in December 1973 and Unit 2 began commercial operation in December 1974.

The PINGP was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria (GDC) for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967.

PINGP has two containment systems, one for each unit. The containments are structurally independent. The discussions of containment system design features and containment system accident response are presented for a single Unit and are equally applicable to either Unit. The primary containment system consists of a steel structure, the containment vessel, and its associated engineered safety features systems. The containment vessel is a "free-standing" steel pressure vessel surrounded by a reinforced concrete shield building. The containment vessel, including all its penetrations, is a low leakage steel shell designed to contain radioactive material that may be released from the reactor core following a design basis LOCA. Additionally, the containment and shield building provide shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The safety design basis for the containment system is that the containment vessel must withstand the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate. The design basis accidents that result in a challenge to containment operability from high pressures and temperatures are a LOCA and a MSLB accident. As discussed in PINGP USAR Section 5.2.1, the containment vessel is designed for a maximum internal pressure of 46 psig and a temperature of 268°F.

Containment response for LOCAs was performed in accordance with the methodology described in PINGP USAR, Appendix K using the CONTEMPT analysis code. Containment response for MSLB accidents was performed utilizing CONTEMPT as described in PINGP USAR Section 14.5.5. Following NRC approval, the amendment would permit NMC, when needed, to update the containment post accident response analyses using more current methodology consistent with methods used by other NMC plants. GOTHIC 7 is a recently developed methodology supported by Westinghouse, which is also used to model containment response to design bases accidents.

The NRC has previously approved use of GOTHIC 7 for the Kewaunee Nuclear Power Plant (KNPP). The NRC safety evaluation, dated September 29, 2003, in the Applicability section stated, ". . . the conclusions of this safety evaluation with respect to GOTHIC 7.0 are applicable only to the Kewaunee Nuclear Power Plant." KNPP is similar to PINGP, in that both are "Westinghouse 2-loop" plants designed by the same architect/engineering firm. Both KNPP and PINGP have a dual containment design, a "free-standing" steel containment shell that is enclosed in a concrete shield building. In addition, the engineered safeguards equipment design is similar. As stated in this application, NMC will apply modeling assumptions and restrictions to the PINGP analyses similar to those approved for KNPP as discussed in Exhibits C and D.

4.0 TECHNICAL ANALYSIS

Current Licensing Basis for Containment Response Analysis

The safety design basis for the containment system is that the containment vessel must withstand the pressures and temperatures of the limiting design basis accident without exceeding the design leakage rate. Two design basis accidents, a LOCA and a MSLB, are evaluated that result in containment temperatures and pressure which challenge the vessel design.

The current analyses of record for containment response to a LOCA were performed in accordance with methodology currently described in Appendix K of the PINGP USAR. The results discussed in Appendix K are based on computer analysis performed using the CONTEMPT code. Pressure and temperature behavior subsequent to the accident was determined by calculations evaluating the combined influence of the energy sources, heat sinks and engineered safety features. The inputs and assumptions used in these analyses regarding heat input (mass and energy release) and heat removal (active and passive) were made to maximize the containment pressure. This methodology dates to when PINGP was originally licensed in the early 1970's.

Analyses for containment response to a MSLB were performed by internal company organizations using a PINGP specific NRC approved methodology as described in PINGP USAR Section 14.5.5. For the MSLB, this methodology also used the CONTEMPT code to model the containment systems and structures.

Proposed Containment Response Analysis

NMC proposes to use the code for Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7) to model PINGP containment response for LOCA and MSLB accidents. The GOTHIC 7 code takes mass and energy inputs provided by another analysis code and models the containment response to calculate resulting containment temperatures and pressures over the duration of the event. These analyses include conservative assumptions for heat removal including a limiting single failure of the engineered safeguards equipment. NMC proposes to use the results of the GOTHIC 7 containment analyses to determine containment pressure and temperature profiles.

Mass and Energy (M&E) Inputs

As discussed in Section 2.7 of WCAP-16219-P (Exhibit C), the LOCA and MSLB transient M&E releases are calculated separately and input to the GOTHIC 7 containment models via boundary conditions.

The M&E input used in the GOTHIC 7 and CONTEMPT LOCA and MSLB benchmarking cases is from the existing PINGP licensing basis LOCA and MSLB containment response, as documented in Appendix K and Section 14.5.5.3.1 of the

PINGP USAR. The M&E input used in the LOCA GOTHIC 7 containment evaluation model sample case was calculated using the Westinghouse method for LOCA M&E, as described in WCAP-10325. The M&E used as input into the MSLB GOTHIC 7 containment evaluation model sample case was calculated using the Westinghouse method for MSLB M&E, as described in WCAP-8822. The M&E release input used in these sample cases is considered representative of the actual expected release inputs for PINGP. The M&E input used in the proposed GOTHIC 7 analyses will continue to use calculation methods that are not a departure from a method of evaluation described in the PINGP USAR.

Basis for Proposed Licensing Basis Change

As discussed in Exhibits C and D, the GOTHIC 7 code modeling assumptions and input values were compared with the CONTEMPT code and differences were identified. To determine the effect of these differences, the GOTHIC 7 model was modified for benchmark comparisons with the original CONTEMPT models. The benchmark comparisons show that the GOTHIC 7 model results were close to those predicted by CONTEMPT.

The GOTHIC 7 model was used to produce sample results for both transients. The results predicted that the containment would remain below maximum design internal pressure (46 psig) and temperature (268°F) for both cases.

As discussed in Exhibits C and D, the benchmark cases support that the GOTHIC 7 code maintains a similar degree of conservatism with respect to the present licensing basis CONTEMPT code. The requested change to allow use of the GOTHIC 7 code is consistent with existing design bases and conforms to safety analysis acceptance criteria. NMC concludes from these results that the proposed licensing basis change to allow the use of GOTHIC 7 is effective for the proposed purpose and will not create a circumstance inimical to safe operation.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The Nuclear Management Company has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No

   The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing use of the Generation of Thermal-Hydraulic

Information for Containment, Version 7.1patch1, to model containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents.

The containment is not an accident initiator, thus changing the containment modeling methodology does not increase the probability of an accident. This license amendment proposes to use a new methodology for modeling containment response analyses following an accident inside containment involving release of steam and water. This amendment does not alter the nuclear reactor core or reactor coolant system equipment, nor does it alter the methods or equipment used directly in mitigation of an accident. Thus radioactive releases inside containment due to an accident and radioactive releases from containment are not affected by the proposed change in analysis methodology. As discussed in Exhibits C and D, the Gothic 7 sample results for the LOCA and MSLB transients predicted that the containment would remain below design pressure for both cases. Therefore, this change does not increase the consequences of an accident previously evaluated.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed amendment will change the Prairie Island Nuclear Generating Plant licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1, to model containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents.

The proposed amendment does not involve changes to plant design, hardware, system operation, or procedures involved with the containment function. The proposed changes include application of new methodology for analysis of containment response following a loss of coolant accident or steam line break accident. The results of the analyses are used to demonstrate that the acceptance criteria for the containment structure continue to be met. These changes do not create the possibility for a new or different kind of accident.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. **Do the proposed changes involve a significant reduction in a margin of safety?**

   Response: No

   The proposed amendment will change the Prairie Island Nuclear Generating Plant (PINGP) licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1 (GOTHIC 7), to model containment response for loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents.

   The proposed licensing basis change to use GOTHIC 7 affects the design basis loss of coolant accident and main steam line break containment accident analyses. As discussed in Exhibits C and D, the Gothic 7 sample results for the LOCA and MSLB transients predicted that the containment would remain below design pressure for both cases. The GOTHIC 7 accuracy in this application has been verified through benchmark analyses against the current analyses of record, validated against recognized standard data, and found to be appropriate for application to the PINGP design basis accidents. Safety analysis acceptance criteria are satisfied and adherence to safety analysis acceptance criteria using GOTHIC 7 assures that Technical Specification limits will not be exceeded during normal operation. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, the Nuclear Management Company concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

General Design Criteria

The construction of the Prairie Island Nuclear Generating Plant was significantly complete prior to issuance of 10 CFR 50 Appendix A General Design Criteria (GDC). The PINGP was designed and constructed to comply with the Atomic Energy Commission General Design Criteria (AEC GDC) as proposed on July 10, 1967 as described in the plant Updated Safety Analysis Report (USAR). AEC GDC 10 and 49 provide design guidance for containment.

Criterion 10 – Containment

> *Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the*

situation requires the functional capability to protect the public.

The proposed amendment will change the PINGP licensing basis by allowing use of the Generation of Thermal-Hydraulic Information for Containment, Version 7.1patch1(GOTHIC 7) to model containment response for design basis accidents, loss of coolant accidents (LOCA) and main steam line break (MSLB) accidents. The proposed amendment does not physically impact the containment structure or the other engineered safety features, which together assure that the containment functional capability is retained. The application of GOTHIC 7 provides assurance that the containment design basis is met. The results predicted by GOTHIC 7 for the design basis accident analyses, a large coolant boundary break and a main steam line break, remain within limiting design basis accidents of record. GOTHIC 7 accuracy in this application has been verified through benchmark analyses against the current analyses of record, validated against recognized standard data, and found to be appropriate for application to the PINGP design basis accidents.

<u>Criterion 49 – Containment Design Basis</u>

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reaction, that could occur as a consequence of failure of emergency core cooling systems.

The design containment leakage rate is the maximum allowable containment leakage rate at the containment design maximum internal pressure. The GOTHIC 7 code is used to determine the largest credible energy release following a LOCA and resulting internal pressure and temperature. By assuring that the containment design maximum internal pressure is not exceeded from the energy release, the previously analyzed containment leakage rate and associated dose calculations are not impacted.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

The Nuclear Management Company has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22(c)(9). Therefore, pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

EXHIBIT B

NOTARIZED AFFIDAVIT FROM WESTINGHOUSE
APRIL 29, 2004

(7 pages to follow)

Westinghouse

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Our ref: CAW-04-1828

April 29, 2004

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-16219-P, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Prairie Island Nuclear Generating Plants," Revision 0, April 2004 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-04-1828 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 2.390 of the Commission's regulations.

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

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-04-1828, and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: G. Shukla
W. Macon
E. Peyton

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

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, 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. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 29th day
of April, 2004

Sharon L. Fiori
Notary Public

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 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 2.390 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 2.390 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 2.390, 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 WCAP-16219-P, "Development and Qualification of a GOTHIC Containment Evaluation Model for the Prairie Island Nuclear Generating Plants," Revision 0, April 2004 (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 and 2 is expected to be applicable for other licensee submittals in response to certain NRC containment licensing analysis requirements.

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

(a) Assist the customer to obtain NRC approval of their GOTHIC containment analysis evaluation model application.

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) 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 calculations 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.

EXHIBIT C

WCAP-16219-P
DEVELOPMENT AND QUALIFICATION OF A GOTHIC CONTAINMENT EVALUATION MODEL FOR THE PRAIRIE ISLAND NUCLEAR GENERATING PLANTS
MARCH 2004

(Proprietary Version)

(137 pages to follow)

EXHIBIT D

WCAP-16219-NP
DEVELOPMENT AND QUALIFICATION OF A GOTHIC CONTAINMENT EVALUATION MODEL FOR THE PRAIRIE ISLAND NUCLEAR GENERATING PLANTS
MARCH 2004

(Non-proprietary Version)

(137 pages to follow)