Indiana Michigan
Power Company
Cook Nuclear Plant
One Cook Place
Bridgman MI 49106
616-465-5901

INDIANA
MICHIGAN
POWER

November 15, 2002

AEP:NRC:2349-02
10 CFR 50.90
10 CFR 2.790

Docket No.: 50-316

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Donald C. Cook Nuclear Plant Unit 2
RESPONSE TO NUCLEAR REGULATORY COMMISSION
REQUESTS FOR ADDITIONAL INFORMATION REGARDING
PROPOSED LICENSE AMENDMENT FOR UNIT 2
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE CURVES,
AND REQUEST FOR EXEMPTION FROM REQUIREMENTS IN 10 CFR 50
(TAC NO. MB5699)

Reference:

1. Letter from J. E. Pollock, Indiana Michigan Power Company (I&M), to U. S. Nuclear Regulatory Commission (NRC) Document Control Desk, "License Amendment Request for Unit 2 Reactor Coolant System Pressure-Temperature Curves, and Request for Exemption from Requirements in 10 CFR 50.60(a) and 10 CFR 50, Appendix G," AEP:NRC:2349-01, dated July 23, 2002

2. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant, Unit 2 – Request for Additional Information Regarding License Amendment Request, 'Reactor Coolant System Pressure – Temperature Curves,' dated July 23, 2002, (TAC No. MB5699)," dated September 27, 2002

This letter provides I&M's response to NRC requests for additional information regarding a proposed license amendment. The proposed license amendment pertains to the reactor coolant system (RCS) pressure-temperature curves in the Donald C. Cook Nuclear Plant (CNP) Unit 2 Technical Specifications (TS).

A001

By Reference 1, I&M proposed to amend Facility Operating License DPR-74, for CNP Unit 2, to revise the RCS pressure-temperature curves in TS Figures 3.4-2 and 3.4-3. On September 27, 2002, the NRC staff requested additional information, via Reference 2, regarding the proposed amendment. The NRC also requested additional information regarding the proposed amendment on October 7, 2002, via telecopy.

Attachment 1 to this letter provides I&M's response to the requests for additional information identified by the NRC on September 27, 2002, and October 7, 2002. Attachment 2 provides specific data requested by the NRC on October 7, 2002. Westinghouse Electric Company LLC ("Westinghouse"), has designated information in Attachment 2 as proprietary pursuant to 10 CFR 2.790. Attachment 3 provides an affidavit from Westinghouse setting forth the basis on which information contained in Attachment 2 may be withheld from public disclosure. Attachment 4 contains a non-proprietary version of Attachment 2. There are no new regulatory commitments made in this letter.

The information provided in this letter consists of supporting information for the amendment request submitted by Reference 1. The information provided in this letter does not alter the requested amendment and does not affect the validity of the original evaluation of significant hazards considerations performed in accordance with 10 CFR 50.92 or the environmental assessment performed in accordance with 10 CFR 51.21 as documented in Enclosure 2 of the referenced letter.

Should you have any questions, please contact Mr. Brian A. McIntyre, Manager of Regulatory Affairs, at (269) 697-5806.

Sincerely,

J. E. Pollock

J. E. Pollock
Site Vice President

JRW/rdw

Attachment:

1. Response to Nuclear Regulatory Commission Requests for Additional Information
2. Westinghouse Letter LTR-EMT-02-298, "Thermal Stress Intensity Factors for D. C. Cook Unit 2 PT Curves (Proprietary Version)," dated October 22, 2002
3. Application for Withholding Proprietary Information from Public Disclosure
4. Westinghouse Letter LTR-EMT-02-297, "Thermal Stress Intensity Factors for D. C. Cook Unit 2 PT Curves (Non-Proprietary Version)," dated October 22, 2002

c: K. D. Curry, Ft. Wayne AEP
J. E. Dyer, NRC Region III
MDEQ - DW & RPD
NRC Resident Inspector
J. F. Stang, Jr., NRC Washington, DC
R. Whale, MPSC

bc: A. C. Bakken III
M. J. Finissi
S. A. Greenlee
D. W. Jenkins, w/o attachments
J. A. Kobyra, w/o attachments
B. A. McIntyre, w/o attachments
J. E. Newmiller
J. E. Pollock
D. J. Poupard
T. Satyan-Sharma/P. G. Schoepf
M. K. Scarpello, w/o attachments
T. K. Woods, w/o attachments

AFFIRMATION

I, Joseph E. Pollock, being duly sworn, state that I am Site Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

JE Pollock

J. E. Pollock
Site Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 15 DAY OF NOVEMBER, 2002

Jennifer L Kernosky
Notary Public

My Commission Expires 5/26/05

JENNIFER L KERNOSKY
Notary Public, Berrien County, Michigan
My Commission Expires May 26, 2005

ATTACHMENT 1 TO AEP:NRC:2349-02

RESPONSE TO NUCLEAR REGULATORY COMMISSION REQUESTS FOR ADDITIONAL INFORMATION

This attachment provides Indiana Michigan Power Company's (I&M) response to Nuclear Regulatory Commission (NRC) requests for additional information regarding a proposed license amendment. The proposed amendment would revise the reactor coolant system (RCS) pressure-temperature curves in the Donald C. Cook Nuclear Plant (CNP) Unit 2 Technical Specifications (TS). The proposed amendment was transmitted by a letter dated July 23, 2002, from J. E. Pollock, I&M, to the NRC Document Control Desk. NRC Questions 1 through 5 below were transmitted by a letter dated September 27, 2002, from J. F. Stang, NRC, to A. C. Bakken III, I&M. NRC Question 6 was provided to I&M on October 7, 2002, via telecopy.

Some of the NRC questions and I&M responses refer to Revision 1 and/or Revision 0 of WCAP-13515. WCAPs are published by Westinghouse Electric Company LLC (Westinghouse). WCAP-13515 is, titled "Analysis of Capsule U from Indiana Michigan Power Company D. C. Cook Unit 2 Reactor Vessel Radiation Surveillance Program." This WCAP documents the Charpy V-Notch testing, tensile testing, and analysis that was performed on surveillance capsule U following its removal from the Unit 2 reactor pressure vessel in 1992, after 8.65 effective full power years (EFPY) of operation. This WCAP also documents an analysis to determine the neutron radiation environment within the reactor pressure vessel, including projections of future neutron exposure. Revision 0 of the WCAP was used to support the existing Unit 2 RCS pressure-temperature curves in the TS, and was provided as an attachment to a letter dated March 12, 1993, from E. E. Fitzpatrick, I&M, to T. E. Murley, NRC. Revision 1 of the WCAP was provided as Attachment 3 to the July 23, 2002, letter transmitting the proposed amendment.

NRC Question 6 refers to WCAP-15047, Revision 2. This WCAP is titled, "D. C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation." This WCAP provides the RCS pressure-temperature curves that were proposed as revisions the existing Unit 2 RCS pressure-temperature curves in the TS, and describes how the curves were developed. These curves are based, in part, on fluence data from the revised analysis of surveillance capsule U documented in WCAP-13515, Revision 1. WCAP-15047, Revision 2, was provided as Attachment 4 to the July 23, 2002, letter transmitting the proposed amendment.

NRC Request for Additional Information

By letter dated July 23, 2002 the Indiana Michigan Power Company, licensee for the D. C. Cook Nuclear Power Plant, Unit 2, submitted information and requested approval for Technical Specification changes to revise the pressure vessel pressure temperature curves. The submittal included WCAP-13515 Rev. 1, the surveillance capsule report for capsule U. In the context of WCAP-13515, Rev. 1, please consider the following questions.

NRC Question 1

What is the physical basis for the calculated peak inside surface E > 1.0 MeV at 32 effective full power years to be lower than the original FERRET Code adjusted value? This plant has a thermal shield and the transport cross sections were changed to ENDF/B-VI which should have increased the original value.

Response to NRC Question 1

The maximum projected fluence (E (energy) > 1.0 MeV) at 32 EFPY reported in Revision 1 of WCAP-13515 (1.625E+19 n/cm^2) is less than the corresponding value reported in Revision 0 of WCAP-13515 (1.71E+19 n/cm^2) due to the introduction of more aggressive low leakage loading patterns. The current fuel management approach tends to reduce the power generation on the core periphery resulting in a reduced fluence rate at the pressure vessel.

Revision 0 of the WCAP included a unit specific evaluation through completion of Fuel Cycle 8. The projections for future operation were based on a core power distribution representative of the average of Fuel Cycles 1 through 8. This average power distribution included several loading patterns with relatively high peripheral power. Revision 1 of the WCAP included a unit specific evaluation through the completion of Fuel Cycle 12, and the projections for future operation were based on a core power distribution representative of the average of Fuel Cycles 10 through 12, with a 10 percent positive bias applied to the peripheral power. The loading patterns for these low leakage cycles are representative of the intended future operation of Unit 2.

A comparison of the methodologies used in the two revisions of the WCAP is achieved by comparing the unit specific fluence analyses at the conclusion of Fuel Cycle 8 (the last cycle analyzed explicitly in Revision 0 of the WCAP).

From Revision 0 of the WCAP, the FERRET adjusted maximum exposure of the pressure vessel wall at the end of Fuel Cycle 8 is listed as:

FERRET Fluence = 4.65E+18 n/cm^2
(Table 6-13 in Revision 0 of the WCAP)

Based on an applied bias factor of 1/0.94 = 1.06 (Table 6-12 in Revision 0 of the WCAP), the corresponding ENDF/B-IV calculated maximum pressure vessel fluence is:

ENDF/B-IV Calculated Fluence = 4.38E+18 n/cm^2

From Revision 1 of the WCAP, the ENDF/B-VI calculated maximum pressure vessel fluence at the end of Fuel Cycle 8 is listed as:

ENDF/B-VI Calculated Fluence = 4.597E+18 n/cm^2
(Table 6-2 in Revision 1 of the WCAP)

From this comparison of the maximum pressure vessel fluence at the end of the Fuel Cycle 8, it can be seen that the ENDF/B-VI calculation is higher than the ENDF/B-IV calculation, and is in excellent agreement with the results of the FERRET adjustment of the ENDF/B-IV data.

NRC Question 2

Is the old and the new FERRET Code the same? (We noted that the calculated value was used and not the FERRET Code adjusted value).

Response to NRC Question 2

The FERRET computer code is a linear least squares adjustment code that has not been changed between the applications documented in Revision 0 and Revision 1 of WCAP-13515. However, due to the evolution from ENDF/B-IV to ENDF/B-VI cross-sections, some of the inputs to the FERRET adjustment procedure have changed.

There are three fundamental inputs to the FERRET adjustment procedure. These are:

- Calculated neutron energy spectrum and associated uncertainties.
- Dosimetry reaction cross-sections and associated uncertainties.
- Measured dosimeter reaction rates and associated uncertainties.

In current evaluations, the calculated neutron energy spectra are based on discrete ordinates transport calculations using ENDF/B-VI cross-sections, and the uncertainties are based on the latest benchmarking comparisons and sensitivity studies. Prior evaluations used calculations based on ENDF/B-IV transport cross-sections. The ENDF/B-VI analyses result in an increase in the magnitude of the calculated spectra and a reduction in the uncertainty associated with the calculations.

The dosimetry reaction cross-sections used in the current least squares analyses are, likewise, based on the latest ENDF/B-VI data, and include extensive uncertainty evaluations. The dosimetry cross-section data set used by Westinghouse is recommended by the American Society for Testing and Materials (ASTM) for light water reactor applications (ASTM E1018-01, "Standard Guide for Application of ASTM Evaluated Cross Section Data File, Matrix E 706 (IIB)," 2002). Prior least squares evaluations used dosimetry cross-sections obtained from the ENDF/B-IV and ENDF/B-V data files.

Measured dosimeter reaction rates have not changed in the Unit 2 analyses.

NRC Question 3

The former plates have been added to the revised analysis. Were any of the dosimeters in the shadow of the former plates? Is the peak location in the shadow of the former plates?

Response to NRC Question 3

As described in the following discussion, four dosimetry sets (top-middle, middle, bottom-middle, and bottom) are in the shadow of the former plates. However, the peak location; i.e., maximum pressure vessel fluence, is not in the shadow of the former plates.

Considering the midplane of the active fuel as Z = 0.0, the elevation of the former plates relative to the core midplane is summarized as follows:

Component	Height [cm]
Core Top	182.88
Former 7	140.90 to 144.39
Former 6	89.69 to 93.19
Former 5	38.49 to 41.98
Core Midplane	0.00
Former 4	-16.38 to -12.88
Former 3	-67.58 to -64.09
Former 2	-118.79 to -115.30
Former 1	-173.65 to -170.16
Core Bottom	-182.88

Relative to the maximum pressure vessel fluences listed in Table 6-14 of WCAP-13515, Revision 1, the fluence values given for 32 and 36 EFPY of operation occur at an elevation of approximately +9 cm (i.e., 9 cm above core midplane). Due to the average axial shape used in the future fluence projections for 48 and 54 EFPY, the location of the maximum fluence shifts to an elevation of approximately -88 cm, i.e., 88 cm below core midplane. Both of these axial elevations are located midway between former plates. Therefore, the peak location is not in the shadow of the former plates.

The surveillance capsules incorporated into the Unit 2 reactor are centered on the core midplane (Z = 0.0) and have a specimen stack height of 99.56 cm. Thus, the capsules span an axial range extending from -49.78 to +49.78 cm relative to the core midplane. From the table above, it can be seen that only Formers 4 and 5 impact this axial span. Former 4 is located below the midplane of the specimen stack and Former 5 is positioned near the top of the stack.

Dosimeters are located within the specimen stack at five axial elevations designated top, top-middle, middle, bottom-middle, and bottom. The following tabulation indicates the axial center of the specimens containing dosimeter wires.

Dosimeter Designation	Center of Dosimetry Set [cm]
Top	44.38
Top-Middle	18.85
Middle	-1.97
Bottom-Middle	-25.49
Bottom	-47.08

From this tabulation, it is noted that the neutron sensors positioned at the top-middle, middle, bottom-middle, and bottom axial elevations are located well away from the formers. The sensors located at the top elevation are positioned near the axial location of Former 5.

In each capsule, the positioning of Former 5 has the potential to impact the measurements obtained with one iron wire, one bare cobalt-aluminum wire, and one cadmium covered cobalt-aluminum wire. Of these, only the iron wire has an impact on fast neutron evaluations. In the Unit 2 application, iron wires are placed at all five axial elevations. An examination of Table 6-8 of WCAP-13515, Revision 1 shows that for all four capsules removed from Unit 2, there is no statistically significant difference between iron measurements at the top location and measurements obtained at the other four axial locations. It can be concluded, therefore, that the presence of Former 5 has a minimal impact on the measurements obtained near the top of the capsule.

NRC Question 4

The γ-fission, U-235 impurity, and Pu-239 built-in corrections (Page 6-8) seem to be new in the revision. How were these corrections derived?

Response to NRC Question 4

The "Page 6-8" noted in the NRC question refers to Page 6-8 of WCAP-13515, Revision 1. The -gamma fission corrections to the U-238 and Np-237 fission dosimeters are now standard practice for all dosimetry evaluations, in accordance with Regulatory Position 2.1.2 of Regulatory Guide 1.190, "Calculation and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001. The corrections were determined for each capsule location from the results of the ENDF/B-VI transport calculations using the BUGLE-96 library. The transport calculations were completed for the entire 67 group structure (47 neutron, 20 gamma ray) included in the BUGLE-96 library. From these calculations, the ratio of gamma ray induced fission to neutron induced fission was obtained for both of the fission sensors. Based on these calculated ratios, the correction factors associated with the Unit 2 capsules were determined as follows.

Capsule ID and Location	Ratio [U-238(γ,f)]/[U-238 (n,f)]	(γ,f) Correction $(1+\text{Ratio})^{-1}$
T (40 Degrees)	0.0439	0.985
Y (40 Degrees)	0.0439	0.985
X (40 Degrees)	0.0439	0.985
U (40 Degrees)	0.0439	0.985

Capsule ID and Location	Ratio [Np-237(γ,f)]/[Np-237(n,f)]	(γ,f) Correction $(1+\text{Ratio})^{-1}$
T (40 Degrees)	0.0156	0.985
Y (40 Degrees)	0.0156	0.985
X (40 Degrees)	0.0156	0.985
U (40 Degrees)	0.0156	0.985

The data in the above tables indicate that the gamma ray induced fission corrections for the Unit 2 fission sensors are approximately 4 percent and 1.5 percent for U-238 and Np-237, respectively.

Additional corrections for trace impurities of U-235 and for the build-in of plutonium isotopes in U-238 fission sensors has always been a part of dosimetry evaluations performed by Westinghouse. Due to the conversion of U-238 to Pu-239 over time, these corrections are a function of the total fluence accrued by the individual sensors. That is, the longer the irradiation, the greater the impact of plutonium fissioning. The corrections used in the Unit 2 dosimetry evaluations were obtained using the ORIGEN code to develop a correlation defining the U-238(n,f) contribution to the total integrated fissions in the dosimeter as a function of the neutron fluence experienced by the sensor. The specific corrections used in the evaluation of the Unit 2 U-238 sensors are summarized as follows:

Capsule ID and Location	Calculated Fluence (E > 1.0 MeV) [n/cm^2]	Fractional U-238 Contribution
T (40 Degrees)	2.38E+18	0.875
Y (40 Degrees)	6.64E+18	0.859
X (40 Degrees)	1.02E+19	0.845
U (40 Degrees)	1.58E+19	0.823

NRC Question 5

It is stated that a 10 percent positive bias was applied to the neutron sources for Cycles 13 and on. Was there also an assumption of low leakage loadings made for the same cycles?

NRC Question 5

It is stated that a 10 percent positive bias was applied to the neutron sources for Cycles 13 and on. Was there also an assumption of low leakage loadings made for the same cycles?

Response to NRC Question 5

As noted in the response to NRC Question 1, the future fluence projections for the Unit 2 reactor vessel were based on a core power distribution representative of the average of Fuel Cycles 10 through 12. All of these cycles were based on the low leakage fuel management concept. Since I&M intends to treat the average core power distribution used in the fluence projections as a guide for future core designs, the 10 percent positive bias was applied in the fluence evaluation to establish a margin for these future designs.

NRC Question 6

The staff requires the following information to complete its review of the license amendment request for the D.C. Cook 2 32 EFPY pressure-temperature (P-T) limit curves that were proposed based on the methods of Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements, Section XI, Division 1." The latest editions of ASME Section XI Appendix G, endorsed by reference in 10 CFR 50.55a, allow for use of plant specific K_{IT} values and temperature gradient values as acceptable inputs for the calculation of P-T limits for operating reactors. For each 32 EFPY P-T limit data point given in Tables 9-1 and 9-2 of Topical Report WCAP-15047, Revision 2, provide the corresponding K_{IT} value and temperature gradient value (i.e., the ΔT values between the temperatures for RCS coolant and those at the 1/4T and 3/4T thickness locations of the RV) that were used for calculation of the data point.

Response to NRC Question 6

The requested data is provided in Attachment 2 to this letter.

SOUTHERN CALIFORNIA
EDISON
An EDISON INTERNATIONAL℠ Company

Dwight E. Nunn
Vice President

November 14, 2002

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Subject: **Docket No. 50-206**
San Onofre Nuclear Generating Station, Unit 1

Gentlemen:

Southern California Edison (SCE) is seeking a Real Estate License from the Camp Pendleton Marine Corps Base to allow SCE to transport the San Onofre Unit 1 (SONGS 1) reactor vessel package across Camp Pendleton land. A conference call was held on May 28, 2002 with personnel from the U. S. Nuclear Regulatory Commission (NRC), U. S. Marine Corps (USMC), and SCE on this subject to discuss the responsibility for conducting an environmental review that will satisfy the requirements of the National Environmental Policy Act (NEPA) for the planned action by USMC officials in issuing the required Real Estate License. The conference call is documented in the enclosure to this letter.

In summary, NRC personnel stated that they do not foresee the need for any approvals or other discretionary action by the NRC for the proposed transportation of the SONGS 1 reactor vessel package across Camp Pendleton property. SCE agrees with this conclusion. The transportation of low-level radioactive waste from a decommissioning facility has already been addressed under the existing generic environmental impact statement (NUREG-0586). The NRC will not be performing a NEPA environmental assessment associated with the reactor vessel package transport and cannot assume the role of lead federal agency for USMC action in issuing a Real Estate License for this purpose.

In order for the USMC to proceed with NEPA compliance in a timely fashion, SCE has been asked to obtain written confirmation from NRC that NRC cannot assume the role of lead agency for the NEPA review of issuing the Real Estate License described above.

Sincerely,

Enclosure: Teleconference Summary

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
D. G. Holland, NRC Project Manager, San Onofre Unit 1
C. C. Osterholtz, NRC Senior Resident Inspector, San Onofre Units 2 & 3

P. O Box 128
San Clemente, CA 92674-0128
949-368-1480
Fax 949-368-1490

TELECONFERENCE SUMMARY

Subject: NEPA Review - Reactor Vessel Package Transport

A teleconference was held on May 28, 2002 to discuss NEPA obligations attending to Marine Corps permission for transport of the reactor vessel package through Base property. In particular, USMC representatives wanted to explore the possibility of the NRC taking the lead agency role in approving the transport of the SONGS 1 reactor vessel package through USMC Camp Pendleton property. Parties to the call were:

NRC
Ann Hodgdon – Office of General Counsel
Mike Masnik – Decommissioning Branch Chief
Drew Holland – Project Manager, SONGS 1

Marine Corps/Camp Pendleton
Curtis Permito - Western Area Counsel Office
Stan Norquist - Base Environmental Office
Mark Anderson - Environmental Specialist

SCE
Nino Mascolo – Sr. Attorney
Maryjane Johnson – Environmental & Facilities Manager
John Todd – Decommissioning Project
Dave Brevig – Manager of External Affairs
Dave Pilmer – Licensing Manager, SONGS 1

The purpose of the call was to discuss responsibilities for federal agencies in meeting NEPA obligations supporting SONGS Reactor Vessel package transport through Camp Pendleton. In prior discussions NRC and Camp Pendleton both seemed to think the other should take the responsibility to act as the lead federal agency. Briefly, their reasoning is the following:

Camp Pendleton -

Unit 1 decommissioning is not regulated by USMC and is not part of their military mission. Their only involvement is issuing the Real Estate License for the transportation through the Base. It is unreasonable for USMC to accept the risk and /or responsibility for the entire transportation activity. NRC authored the generic EIS (and supplement) and bears responsibility for site-specific addenda necessary to perform decommissioning. They see radwaste disposal as a significant federal action authorized by NRC.

TELECONFERENCE SUMMARY

NRC -

RV transport is part of decommissioning and has already been authorized through NRC regulations and issuance of the Unit 1 license. No further NRC action is required and no need for additional environmental review. In the absence of a specific discretionary action by the NRC (issuance of license amendment etc.) or unique circumstances at a site that are outside the scope of the generic environmental impact statement (EIS), no site-specific environmental assessment is necessary.

Since NRC licenses certain radwaste shipping packages (for quantities greater than Type A), Mr. Masnik agreed to talk to NRC experts in that field to see if NRC action is anticipated that could serve as the reason for NRC to conduct an environmental review. Ms. Hodgdon referred Camp Pendleton to 10 CFR § 51.22(c)(13) as possibly serving as the basis for a categorical exclusion for environmental consideration in the Real Estate License. Mr. Masnik also explained that the Decommissioning EIS is under revision and he will explore adding a statement about non-radiological environmental effects of radwaste shipping and transportation to the Supplement.

Based on this conference call, there seems to be little interest and no possibility of the NRC taking the lead agency position for the SONGS 1 reactor vessel package transport or to take charge of the NEPA review associated with the transport.

ATTACHMENT 3 TO AEP:NRC:2349-02

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION
FROM PUBLIC DISCLOSURE

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-5282
Direct fax: (412) 374-4011
e-mail: Sepp1ha@westinghouse.com

Attention: Mr. Samuel J. Collins

Our ref: CAW-02-1563

October 25, 2002

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-EMT-02-298, "Thermal Stress Intensity Factors for D. C. Cook Unit 2 PT Curves (Proprietary Version)", October 2002.

Dear Mr. Collins:

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-02-1563 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 American Electric Power Company.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-02-1563 and should be addressed to the undersigned.

Very truly yours,

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Enclosures

cc: G. Shukla/NRR

CAW-02-1563

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, 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:

KAY E. GONGAWARE COMMONWEALTH OF PENNSYLVANIA NOTARY PUBLIC

H. A. Sepp, Manager
Regulatory and Licensing Engineering

Sworn to and subscribed
before me this 25th day
of October, 2002

Notary Public

Notarial Seal
Kay E. Gongaware, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Feb. 7, 2005
Member, Pennsylvania Association of Notaries

/cm/0253S doc

(1) I am Manager, Regulatory and Licensing Engineering, 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 the Westinghouse Electric Company LLC.

(2) I am making this Affidavit in conformance with the provisions of 10CFR 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 the Westinghouse Electric Company LLC 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 which 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 10CFR 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 letter LTR-EMT-02-298 (Proprietary), October 2002 for D. C. Cook Unit 2 being transmitted by the American Electric Company letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk, Attention Mr. Samuel J. Collins. The proprietary information as submitted for use by American Electric Company for D. C. Cook Unit 2 is expected to be applicable in other licensee submittals in response to certain NRC requests for information to support the Pressure-Temperature curve calculations for D. C. Cook Unit 2.

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

(a) Justify the use of plant-specific thermal stress intensity factors for the Pressure-Temperature curve calculations.

(b) Assist the customer to respond to NRC requests for information.

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 justification for the use of plant-specific thermal stress intensity factors for the Pressure-Temperature curve calculations.

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.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

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ATTACHMENT 4 TO AEP:NRC:2349-02

WESTINGHOUSE LETTER LTR-EMT-02-297, DATED OCTOBER 22, 2002
"THERMAL STRESS INTENSITY FACTORS FOR DONALD C. COOK NUCLEAR PLANT
UNIT 2 PT CURVES (NON-PROPRIETARY VERSION)"

Westinghouse

To: D.W. Sklarsky
cc: J. A. Gresham

Date 10/22/02

From: T. J Laubham
Ext: Win 284-6788
Fax: Win 284-6647

Your ref:
Our ref: LTR-EMT-02-297

Subject: **Thermal Stress Intensity Factors for D.C. Cook Unit 2 PT Curves (Non-Proprietary Version)**

In response to a request for additional information (RAI) from the NRC on pressure-temperature (PT) limit curves, AEP requested Westinghouse supply them with the thermal stress intensity factors associated with the 32 EFPY PT limit curves from WCAP-15047, Revision 2. Attached for AEP's use are the thermal stress intensity factors in question. Table 1 contains the 1/4T and 3/4T thermal stress intensity factors for the 60°F/hr heatup curve, while Table 2 contains the 1/4T thermal stress intensity factors for all the cooldown curves (20, 40, 60 and 100°F/hr). Note that the Cooldown is only limited at the 1/4T location, thus the 3/4T values are not supplied. The heatup curves are limited at both the 1/4T and 3/4T locations, depending on the temperature.

If you have any questions or need additional information, please contact the undersigned.

Author:

T. J. Laubham[1]
Engineering and Materials Technology

Approved by:

J. H. Ledger[1]
Engineering and Materials Technology

Attachments

[1]Official record electronically approved in EDMS 2000

Our ref. LTR-EMT-02-297
10/22/02

Table 1
Kit Values for 60°F/hr Heatup Curve (32 EFPY)

a,b,c

Water Temp (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
(Values up to 245°F are limited by the 3/4T location)		
60		
65		
70		
75		
80		
85		
90		
95		
100		
105		
110		
115		
120		
125		
130		
135		
140		
145		
150		
155		
160		
165		
170		
175		
180		
185		
190		
195		
200		
205		
210		
215		

Table 1 (*Continued*)
Kit Values for 60°F/hr Heatup Curve (32 EFPY)

a,b,c

Water Temp (°F)	1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	3/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
220		
225		
230		
235		
240		
245		
Values above 250°F are limited by the 1/4T location.		
250		
255		
260		
265		
270		
275		
280		
285		
290		

- **Note that the Vessel Radius to the ¼T and ¾T Locations are as follows:**

  1/4T Radius = 88.844" &

  3/4T Radius = 93.094"

Table 2
Kit Values for all Cooldown Curves (32 EFPY)

a,b,c

Water Temp. (°F)	20°F/hr Cooldown 1/4 T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	40°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	60°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)	100°F/hr Cooldown 1/4T Thermal Stress Intensity Factor (KSI SQ. RT. IN.)
215				
210				
205				
200				
195				
190				
185				
180				
175				
170				
165				
160				
155				
150				
145				
140				
135				
130				
125				
120				
115				
110				
105				
100				
95				
90				
85				
80				
75				
70				
65				
60				

(*) Values above 215°F are limited by Steady State.
(**) Values above 210°F are limited by lower rate or Steady State.
(***) Values above 205°F are limited by lower rate or Steady State.
(****) Values above 200°F are limited by lower rate or Steady State.