Progress Energy

Christopher L. Burton
Vice President
Harris Nuclear Plant
Progress Energy Carolinas, Inc.

DEC 09 2010

Serial: HNP-10-105
10 CFR 50.90

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT TO INCORPORATE A REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT METHODOLOGY INTO THE CORE OPERATING LIMITS REPORT (TAC NO. ME3569)

References:
1. Letter from M. Vaaler, Nuclear Regulatory Commission, to C. L. Burton, "Shearon Harris Nuclear Power Plant, Unit 1 – Request for Additional Information Regarding Amendment to Incorporate a Realistic Large Break Loss of Coolant Accident Methodology into the Core Operating Limits Report (TAC NO. ME3569), sent via email dated September 23, 2010

2. Letter from C. L. Burton to the Nuclear Regulatory Commission (Serial: HNP-10-029), "Application for Revision to Technical Specification Core Operating Limits Report (COLR) References for Realistic Large Break LOCA Analysis," dated March 23, 2010 (ML100890596)

Ladies and Gentlemen:

On September 23, 2010, the Harris Nuclear Plant (HNP) received a request from the NRC (Reference 1) for additional information needed to facilitate the review of the License Amendment Request to revise Technical Specification (TS) Section 6.9.1.6 to add the NRC approved topical report (TR) EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors," to its Core Operating Limits Report methodologies list. This original request was submitted as Serial: HNP-10-029 (Reference 2).

Enclosure 1 to this submittal contains HNP's response to the NRC's request for additional information. In order to incorporate the additional changes resulting from this response, HNP proposes to implement this License Amendment prior to Cycle 18 startup, currently scheduled for May 2012. The initial License Amendment Request (Reference 2) identified implementation as within 60 days of NRC approval.

Enclosure 2 contains the marked up and retyped Technical Specification pages, which replace Attachments 1 and 2 of Enclosure 1 to SERIAL: HNP-10-029 (Reference 2) in their entirety.

P.O. Box 165
New Hill, NC 27562

T> 919.362.2502
F> 919.362.2095

A002
NRR

Enclosure 3 contains the associated Regulatory Commitment.

In accordance with 10 CFR 50.91(b), HNP is providing the state of North Carolina with a copy of this response.

Please refer any questions regarding this submittal to Mr. John Caves, Supervisor – HNP Licensing/Regulatory Programs, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct. Executed on
DEC 09 2010

Sincerely,

Christopher L. Burton

CLB/kms

Enclosures:
1. Response to Request for Additional Information
2. Updated Technical Specification Pages
3. Regulatory Commitment
4. Affidavit for Withholding of Proprietary Information
5. Non-Proprietary Enclosure 1

cc:
Mr. J. D. Austin, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, Section Chief N. C. DENR
Mrs. B. L. Mozafari, NRC Project Manager, HNP
Mr. L. A. Reyes, NRC Regional Administrator, Region II

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
UPDATED TECHNICAL SPECIFICATION PAGES

(11 Pages)

Note: this Enclosure replaces Attachments 1 and 2 of Enclosure 1 to SERIAL: HNP-10-029 in their entirety.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
UPDATED TECHNICAL SPECIFICATION PAGES

TECHNICAL SPECIFICATION PAGE MARKUPS
(5 Pages)

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor, $F_Q^{RTP}$, K(Z), and V(Z) for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

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

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

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

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

"cloud" indicates deletions

"cloud" indicates deletions

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

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

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

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

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

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

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

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

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

insert "A"

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. The report shall include:

a. The scope of inspections performed on each SG.

b. Active degradation mechanisms found.

c. Nondestructive examination techniques utilized for each degradation mechanism.

Insert "A":

p. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
UPDATED TECHNICAL SPECIFICATION PAGES

RETYPED TECHNICAL SPECIFICATION PAGES
(4 Pages)

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d. Control Bank Insertion Limits for Specification 3/4.1.3.6,

e. Axial Flux Difference Limits for Specification 3/4.2.1,

f. Heat Flux Hot Channel Factor, $F_Q^{RTP}$, K(Z), and V(Z) for Specification 3/4.2.2,

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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f. DELETED.

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

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

p. EMF-2103(P)(A),"Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

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6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT TO INCORPORATE A REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT METHODOLOGY INTO THE CORE OPERATING LIMITS REPORT

REGULATORY COMMITMENT

The action in this document committed to by Harris Nuclear Plant (HNP) regarding the March 28, 2010, 10 CFR 50.90 License Amendment Request (LAR), "Application for Revision to Technical Specification Core Operating Limits Report (COLR) References for Realistic Large Break LOCA Analysis," is identified in the following table. Statements in this submittal, with the exception of that in the table below, are provided for information purposes and are not considered commitments. Please direct any questions regarding this document or any associated regulatory commitments to the Supervisor, Licensing/Regulatory Programs.

| Item | Commitment | Completion Date |
|---|---|---|
| 1 | The LOCA analysis performed for the Harris Nuclear Plant will adhere to the deviations from EMF-2103, Rev 0 noted in Section 1 of ANP-2853(P) until such time as:<br><br>a. AREVA develops a new revision of EMF-2103,<br>b. The NRC approves the new revision of EMF-2103, and<br>c. HNP implements the new, NRC-approved revision of EMF-2103. | This commitment will terminate when a, b, and c are met or when HNP obtains NRC approval of an alternate replacement Large Break LOCA methodology. |

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING
AMENDMENT TO INCORPORATE A REALISTIC LARGE BREAK LOSS OF COOLANT
ACCIDENT METHODOLOGY INTO THE CORE OPERATING LIMITS REPORT

AREVA
AFFIDAVIT PURSUANT TO 10 CFR 2.390
(3 Pages)

# AFFIDAVIT

COMMONWEALTH OF VIRGINIA )<br>
) ss.<br>
CITY OF LYNCHBURG )

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in the attachment to a letter from Christopher L. Burton, (Vice President, Harris Nuclear Plant) to Document Control Desk (NRC) entitled "Response to Request for Additional Information Regarding Amendment to Incorporate a Realistic Large Break Loss of Coolant Accident Methodology into the Core Operating Limits Report (TAC No. ME3569)," Serial HNP-10-105 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 28th day of October, 2010.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

Enclosure 1 (Non-Proprietary)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/RENEWED LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING AMENDMENT TO INCORPORATE A REALISTIC LARGE BREAK LOSS OF COOLANT ACCIDENT METHODOLOGY INTO THE CORE OPERATING LIMITS REPORT

Summary

By letter dated March 23, 2010, (ADAMS) Accession No. ML100890596, Carolina Power & Light Company, now doing business as Progress Energy Carolinas, Inc., submitted a proposed amendment for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed amendment would revise Technical Specification (TS) Section 6.9.1.6 to add the Nuclear Regulatory Commission (NRC) approved topical report (TR) EMF-2103(P)(A), Revision 0, "Realistic Large Break LOCA [Loss of Coolant Accident] Methodology for Pressurized Water Reactors," to the Core Operating Limits Report methodologies list.

This change will allow the use of thermal-hydraulic computer analysis code S-RELAP5 for the Final Safety Analysis Report (FSAR) Chapter 15 realistic large break LOCA in the HNP safety analyses. TR EMF-2103(P)(A), Revision 0, was approved by the NRC on April 9, 2003, for the application of the S-RELAP5 thermal-hydraulic analysis computer code to FSAR Chapter 15 realistic large break LOCA.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the information submitted by the licensee, and based on this review determined the following information is required to complete the evaluation of the subject amendment request:

**Request 1:**
The subject license amendment request (LAR) proposes to add topical report (TR) EMF-2103(P)(A), "Realistic Large Break LOCA [Loss of Coolant Accident] Methodology for Pressurized Water Reactors," to the Core Operating Limits Report (COLR) references list in the technical specifications (TS), while retaining reference to a previous/legacy method. Please revise the proposed TS pages to remove reference to the legacy method and revise the requested implementation date to allow timely implementation of the change without a need to retain the additional reference.

**Response:**
The proposed TS pages have been modified to reflect the deletion of EMF-2087(P)(A), “SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications,” (current TS 6.9.1.6.f) and the addition of EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," as TS 6.9.1.6.p. The revised implementation date, prior to Cycle 18 startup, will accommodate orderly transition to the new method.

**Request 2:**
As noted in Section 1 of ANP-2853(P), “Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Summary Report,” deviations from the approved realistic large break (RLBLOCA) evaluation model (EMF-2103(P)(A), Revision 0) are necessary to demonstrate compliance with

the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors."

Accordingly, please provide a commitment to adhere to the deviations noted in Section 1 of ANP-2853(P) until such time as:

a. AREVA develops a new revision of EMF-2103,

b. The NRC approves the new revision of EMF-2103, and

c. HNP implements the new, NRC-approved revision of EMF-2103.

The commitment should include language which indicates that meeting conditions a, b, and c, above, or submitting a LAR to implement a different RLBLOCA evaluation method, will obviate the need for this commitment.

**Response:**
Progress Energy commits to the following restrictions on the use of EMF-2103, Rev 0. The commitment is as follows:

> The LOCA analysis performed for the Harris Nuclear Plant will adhere to the deviations from EMF-2103, Rev 0 noted in Section 1 of ANP-2853(P) until such time as:
>
> a. AREVA develops a new revision of EMF-2103,
> b. The NRC approves the new revision of EMF-2103, and
> c. HNP implements the new, NRC-approved revision of EMF-2103.

This commitment (Enclosure 3 of this submittal) will terminate when a, b, and c are met or when HNP obtains NRC approval of an alternate replacement Large Break LOCA methodology.

**Request 3:**
Please review the methodology references listed on revised TS Page 6.9.1.6, and determine the currency of all references listed. Provide the results of this review. Should any references be determined to be inapplicable to current or recent fuel cycles, please revise the TS page to remove them and renumber the reference list accordingly.

**Response:**
The following table lists the analytical methods used to determine the core operating limits as contained in TS 6.9.1.6.2.a through 6.9.1.6.2.o (pages 6-24 through 6-24c), including the TS changes associated with the License Amendment Request dated July 21, 2009. Based on the

information provided in the Cycle 17 COLR, as submitted October 22, 2010, via HNP-10-106 (ML103010147), two methods are not applicable for the current operating cycle:

a. Although XN-75-27 (TS 6.9.1.6.2.a) is not used for determination of the Cycle 17 core operating limits, Supplement 5 of XN-75-27, which describes a method to perform RCCA reactivity measurement using the AREVA rod swap method, is used by HNP for low power physics testing. No change will be made to the TS for this item.

   HNP utilizes EMF-96-029 (TS 6.9.1.6.2.l), rather than XN-75-27, for determining neutronics input to safety during reload design.

b. EMF-2328 (TS 6.9.1.6.2.m), an S-RELAP5 based Small Break LOCA methodology, was a previous addition to TS section 6.9.1.6.2 in preparation for the transfer of all the safety analyses methodologies to the S-RELAP5 code platform. Although it was not used in the determination of the Cycle 17 core operating limits, it will be retained for Cycle 18 use.

Note: ANF 89-151 (TS 6.9.1.6.2.b) performs a similar role as EMF-2310, currently pending NRC approval for inclusion in the HNP TS (License Amendment Request dated July 21, 2009). When individual events are redone using EMF-2310, ANF 89-151 will no longer be used for the specific event and will be removed from the methodology list.

HNP is also proposing removal of the individual "Methodology for Specification..." currently listed under the title of each methodology in TS sections 6.9.1.6.2.a through 6.9.1.6.2.o. While this information will remain in HNP's COLR, removal from the TS is in accordance with Westinghouse Standard Technical Specifications (NUREG-1431), Section 5.6.5.

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| TS 6.9.1.6 (sub item) | Subject | Topical Report | Cycle 17 COLR applicability | Comments |
|---|---|---|---|---|
| a. | Neutronics Design Method | XN-75-27 | No | Used for RCCA reactivity measurement |
| b.[1] | Non-LOCA (ANF-RELAP) | ANF-89-151 | Yes | To be replaced with EMF-2310 (LAR in review) [1] |
| c. | T-H Mixed Core | XN-NF-82-21 | Yes | |
| d. | Fuel Rod Bow | XN-75-32 | Yes | |
| e. | Non-LOCA (MSLB) | EMF-84-093 | Yes | |
| f.[2] | Large Break LOCA- App K | EMF-2087 | Yes | To be replaced with EMF-2103 (this submittal) [2] |
| g | Rod Ejection | XN-NF-78-44 | Yes | |
| h. | Power Distribution (PDC-3) | ANF-88-054 | Yes | |
| i. | Statistical Setpoint - W | EMF-92-081 | Yes | |
| j. | DNB Correlation | EMF-92-153 | Yes | |
| k.[3] | SBLOCA | XN-NF-82-49 | Yes | To be replaced with EMF-2328 (M5 LAR) [3] |
| l. | Neutronics Design Method | EMF-96-029 | Yes | |
| m.[3] | SBLOCA | EMF-2328 | No | Will replace XN-NF-82-49 (M5 LAR) [3] |
| n.[1] | Non-LOCA | EMF-2310 | N/A | Replaces ANF-89-151 (LAR in review) [1] |
| o. | Fuel Mechanical | XN-NF-81-58<br>ANF-81-58<br>XN-NF-82-06<br>ANF-88-133<br>XN-NF-85-92<br>EMF-92-116 | Yes | |

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[1] The addition of EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", is contained in HNP's License Amendment Request submitted July 21, 2009, via HNP-09-055 (ML092150053). Subsequently, HNP-10-073, submitted July 28, 2010 (ML102160371), contained a Regulatory Commitment regarding the use of EMF-2310 (TS 6.9.1.6.2.n), rather than ANF-89-151 methodology, post-implementation.

[2] Implementation of this current LAR will include deletion of EMF-2087 and addition of EMF-2103 to the TS.

[3] HNP intends to submit a LAR to allow the use of M5 cladding. XN-NF-82-49 (TS 6.9.1.6.2.k) will be deleted upon implementation of this future M5 cladding amendment. The replacement methodology, EMF-2328, already approved for use at HNP, is listed as TS 6.9.1.6.2.m.

**Request 4:**
Please provide additional information about the management of the fuel thermal conductivity degradation issue identified in NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation." Specifically:

a. ANP-2853(P), Page 1-3, states:

> For each specific time in cycle, the fuel conditions are computed using RODEX3A prior to starting the S-RELAP5 portion of the analysis. A steady state condition for the given time in cycle using S-RELAP5 is established. A base fuel centerline temperature is established in this process. Then two-transformation adjustment to the base fuel centerline temperature is computed. The first transformation is a linear adjustment for an exposure of 10 MWd/MTU or higher. In the new process, a polynomial transformation is used in the first transformation instead of a linear transformation.

Please clarify the following:

1) Explain how the fuel pellet radial temperature profile is computed.

2) Explain which code is used to calculate this profile, both for initial conditions and through the postulated accident.

3) Explain whether the polynomial transformation is applied merely to the centerline temperature, or to the entire pellet temperature.

b. Provide additional information to describe the polynomial transformation. In particular,

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summarize the data used to develop the polynomial transformation and discuss the consideration of applicable uncertainties.

**Response:**

a. 1) The RODEX3 topical report, ANF-90-145(P)(A), Appendix B, details the calculation of the radial temperature distribution. Attachment A of this Enclosure contains these pages.

2) The RODEX3A fuel model has been incorporated into the S-RELAP5 code to calculate fuel response for transient analyses. The S-RELAP5/RODEX3A model does not calculate the burnup response of the fuel. Instead, fuel conditions at the burnup of interest are transferred via a binary data file from RODEX3A to S-RELAP5, establishing the initial state of the fuel prior to the transient. The data transferred from RODEX3A describes the fuel at zero power. A steady-state S-RELAP5 calculation is required to establish the fuel state at power. The transient fuel pellet radial temperature profile is computed by solving the conduction equation of S-RELAP5. Material properties are taken from RODEX3A and incorporated into S-RELAP5.

3) The adjustment is applied to the entire fuel pellet. The polynomial transformation provides a bias adjustment to the fuel centerline temperature. A sampled parameter provides a random assessment and adjustment of the centerline temperature uncertainty. These are combined and the total adjustment is achieved by iterating a multiplicative adjustment to the fuel thermal conductivity until the desired fuel centerline temperature is reached. Thus, the adjustment is applied to the entire pellet but with variance according to the nodal pellet temperature and the distance from the node to the pellet surface.

b. <u>Original:</u>
The first transformation applies a linear adjustment if the analysis is being performed for fuel which has an exposure of 10 MWd/MtU or higher.

[ ]

where:

$T_{new}$ = New fuel centerline temperature (°F)
B = Burnup (Gwd/MtU or MWd/KgU) and
$T_{original}$ = Base fuel centerline temperature (°F)

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2nd Transformation: Adds a value determined from a random sampling range of [ ] from a Gaussian distribution.

Revised:
1st Transformation: Instead of adding the linear transformation after 10 Mwd/MtU, a different form of correction factor should be applied.

[ ]

where:

$T_{new}$ = New fuel centerline temperature (K)
B = Burnup (Gwd/MtU or Mwd/KgU) and
$T_{original}$ = Base fuel centerline temperature (K).

2nd Transformation: Remains the same as the original method.

The justification for this process comes from analyzing the fuel rod database used for the development of RODEX4. A calculation was created that used RODEX3A to compute fuel centerline temperatures using all the points in the RODEX4 database (EMF-2994(P)). Three cases (cases 432R2, 432R6, and 597R8) were not used from the RODEX4 database. Case 597R8 was not needed for the present application. Cases 432R2 and 432R6 were rod studies that were not configured in a manner which are to be used in these types of comparisons.

The fractional difference between the RODEX3A calculated results and the data in the RODEX4 database was calculated. The temperature fraction for each point in the database was computed as follows:

$$T_{fraction} = \frac{T_{rodex3A} - T_{data}}{T_{rodex3A}}$$

where:

$T_{fraction}$ = Delta fractional temperature of computed to data (K)
$T_{rodex3A}$ = Temperature computed by RODEX3A (K) and
$T_{data}$ = Temperature from the RODEX4 database (K).

A polynomial curve fit was generated from this data set. Figure 1 is the plot of this data and the curve fit.

**Figure 1** (AREVA NP proprietary and confidential)

The curve fit was then inverted about the zero axis. This new polynomial correction is applied regardless of fuel exposure. Figure 2 shows how the new correction factor changes the results. The data for this plot were created by subtracting $T_{rodex3A}$ from $T_{data}$ as a function of burnup.

( )

**Figure 2** (AREVA NP proprietary and confidential)

The new fuel centerline temperatures no longer have a bias off of the zero error line. The approach to use a fractional based correction algorithm was requested by the NRC. Based on the plot of $T_{rodex}$ - $T_{data}$, the uncertainty used in the original basis does not need to be altered. No specific temperature bias is identified in the uncertainty of the data. Therefore, retaining the current Gaussian distribution sampled from [ ] is acceptable.

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**Request 5:**
HNP TS 3.4.3 permits plant operation with a pressurizer level of up to 92 percent of the indicated span, yet the proposed analyzed range extends only from 53.25 to 66.75 percent. Limiting conditions for operation must be established for design features that are initial conditions of design basis accident or transient analyses (per 10 CFR 50.36(c)(ii)(B)). As such, please explain the selection of a limited pressurizer level span in light of the significantly greater range allowed by the current plant TS.

**Response:**
The HNP TS were developed in accordance with the Westinghouse Standard Technical Specifications in use at the time the original operating license was issued. As described in the HNP Technical Specification Bases (3/4.4.3):

> *The limit on the maximum water level in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water level also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.*

The current Standard Technical Specifications, NUREG-1431, Rev. 3.0, does not provide additional information on this purpose. Per the pressurizer Bases Section, B 3.4.9, Applicable Safety Analyses:

> *In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer... The maximum pressurizer water level limit, which ensures that a steam bubble exists in the pressurizer, satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).*

The suggested value is the equivalent to the high pressurizer water level reactor trip. This does not impact the reasonableness of allowing pressurizer level to be a sampled parameter for the methodology and does not make it essential that the sampled band be otherwise controlled by specific LCO requirements. Therefore, there is no need to identify additional LCO's for pressurizer level.

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**Request 6:**
Please provide additional information in order to facilitate a comparison between the assumed accumulator liquid volume expressed in cubic feet, and the TS-controlled volume, which is given in percent indicated level.

**Response:**
The relationship between accumulator volume and % indicated volume is as follows:

| Item | Value (% indicated) | Cubic Feet ($ft^3$) |
|---|---|---|
| Maximum TS SI accumulator level | 96 | 1029.4 |
| Minimum TS SI Accumulator level | 66 | 994.6 |

**Request 7:**
Please provide the radial pin powers surrounding the hot rod for the flattest power distribution postulated during the cycle. Also, please provide the assembly hot rod peak clad temperature (PCT) and the average clad temperature rod in the assembly at the time of PCT.

**Request 8:**
Please explain how the flattest radial power distribution is located for use in demonstrating/verifying that the thermal rod-to-rod radiation contribution from the plant specific hot rod thermal radiation is bounded by the FLECHT thermal radiation heat transfer rate.

**Response (combined):**
The AREVA Realistic large break LOCA (RLBLOCA) Revision 0 evaluation model (EM) does not include a rod-to-rod radiation model. Instead, the convective heat transfer mechanisms are benchmarked and biased to reproduce the results of experiments that do involve rod-to-rod radiation. This approach subsumes rod-to-rod radiation under the convective heat transfer terms. The assessment of this approach presented in ANP-2853P was originally presented as a generic sensitivity study to demonstrate that the approach was reasonable for light water reactors of the US fleet.

For the conditions presented, the study showed that rod-to-rod radiation heat transfer expected from an explicit modeling of rod-to-rod radiation in the plant would be substantially higher than that present in the experiments to which the Revision 0 reflood heat transfer model was benchmarked. The conclusion drawn was not intended to be a specific application for HNP, but to make the case that the Revision 0 approach was reasonable and likely somewhat conservative relative to explicit radiation modeling.

An important factor in the study was the local peaking difference between the hot rod and the surrounding 24 rods. Based on pin assessments for a relatively flat core (plant 6 in Table 4.1 of ANP-2853P), this was set at 4 percent. Further, the calculation conservatively assumed no credit for the assignment of power uncertainty to the hot pin. The relative expectation for the near region local peaking for HNP can be obtained by comparing the local peaking times the assigned uncertainty for plant 6 to the same product for HNP. The assembly specific uncertainty applied for HNP is 4 percent. The radial local peaking is 1.04. This gives a combined value of 1.08 percent compared to the 1.10 for case 6. It is, therefore, reasonable to expect a slightly reduced hot rod offset for HNP but not so much as to jeopardize the conclusion that the Revision 0 reflood heat transfer approach is reasonable to conservative.

Table 3-5 in ANP-2853(P) provided the hot rod PCT and is repeated below for convenience.

| Case # 5 | UO2 Rod |
|---|---|
| PCT | |
| Temperature | 1930°F |
| Time | 132.6 s |
| Elevation | 10.043 ft |
| Metal-Water Reaction | |
| Percent Oxidation Maximum | 1.9498 |
| Percent Total Oxidation | 0.0605 |

The following table summarizes the PCT for all hot rods and average rods at the time of PCT.

| Case Number | UO2 Rod | 2.0% GAD Rod | 4.0% GAD Rod | 6.0% GAD Rod | 8.0% GAD Rod | Hot Assembly | Surrounding 6 Assemblies | Average Core | Peripheral Core |
|---|---|---|---|---|---|---|---|---|---|
| 5 | 1930 | 1926 | 1922 | 1918 | 1913 | 1903 | 1422 | 1414 | 784 |

**Request 9:**
Please identify the limiting single failure applied to the evaluation of downcomer boiling. Please provide the clad temperature versus time for the worst case downcomer boiling simulation.

**Response:**
Section 4.5 in ANP-2853(P) discusses downcomer boiling in regards to the adequacy of the S-RELAP5 modeling, the wall heat release rate, and downcomer fluid distribution. Loss of one train of ECCS is the limiting single failure prescribed by the RLBLOCA evaluation model (EMF-2103) and, accordingly, is used in all the RLBLOCA transients, including that dealing with downcomer boiling.

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The PCT plots, independent of core elevation, are shown in Figures 3 - 8 for all cases in the HNP RLBLOCA analysis. Several cases indicate signs of the downcomer boiling phenomena between 250 – 400 seconds and exhibit a minor amount of secondary heat up. Case 17 (Figure 4) is the closest case to the PCT case that shows signs of downcomer boiling. The maximum clad temperature for case 17, 1806 °F, is 124 °F less than Case 5 (1930 °F), the PCT case. Additionally, for Case 17, the maximum clad temperature associated with the downcomer boiling portion of the transient is ~70 °F below its maximum clad temperature (1806 °F). Therefore, downcomer boiling plays no role in defining the PCT for the HNP RLBLOCA analysis.

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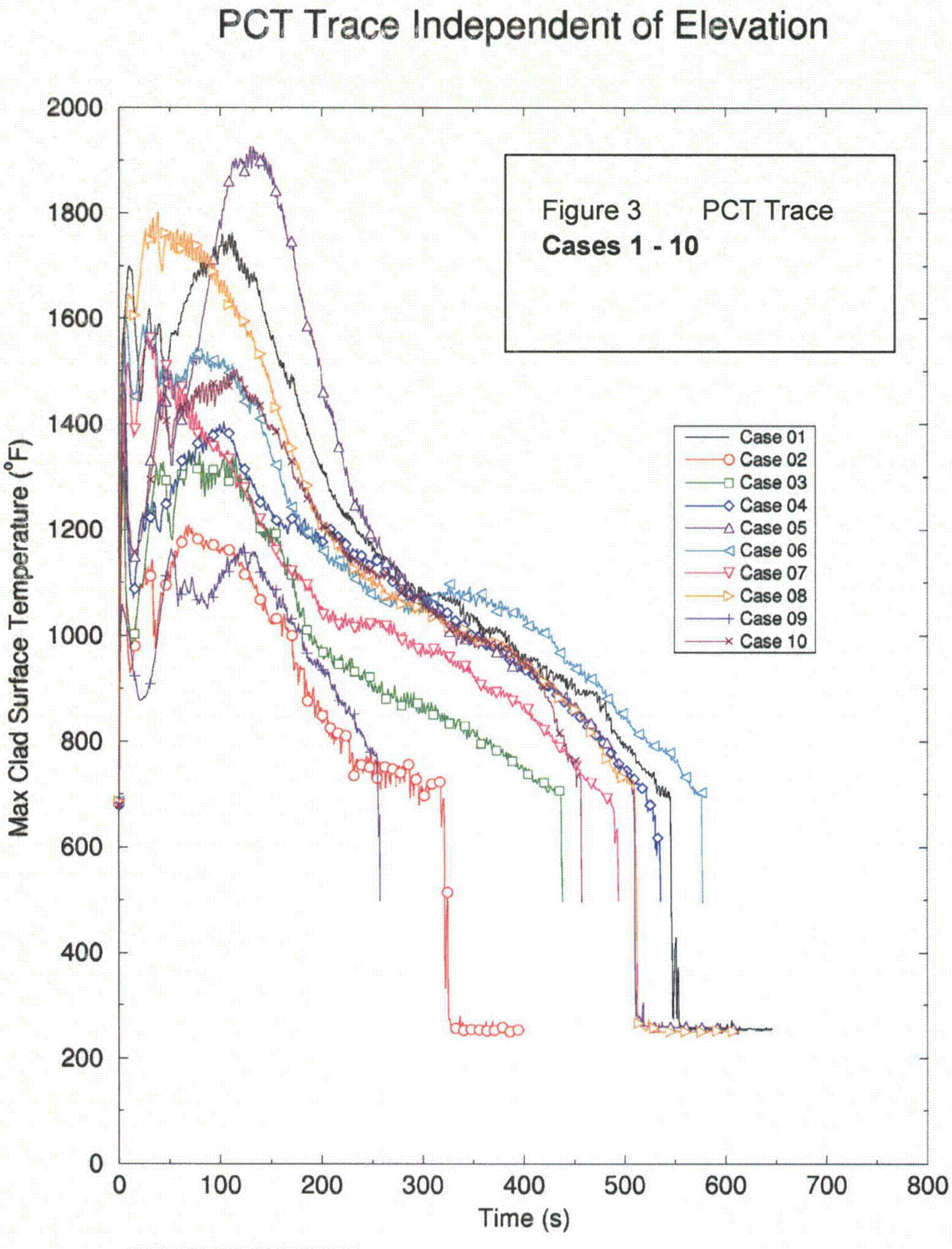

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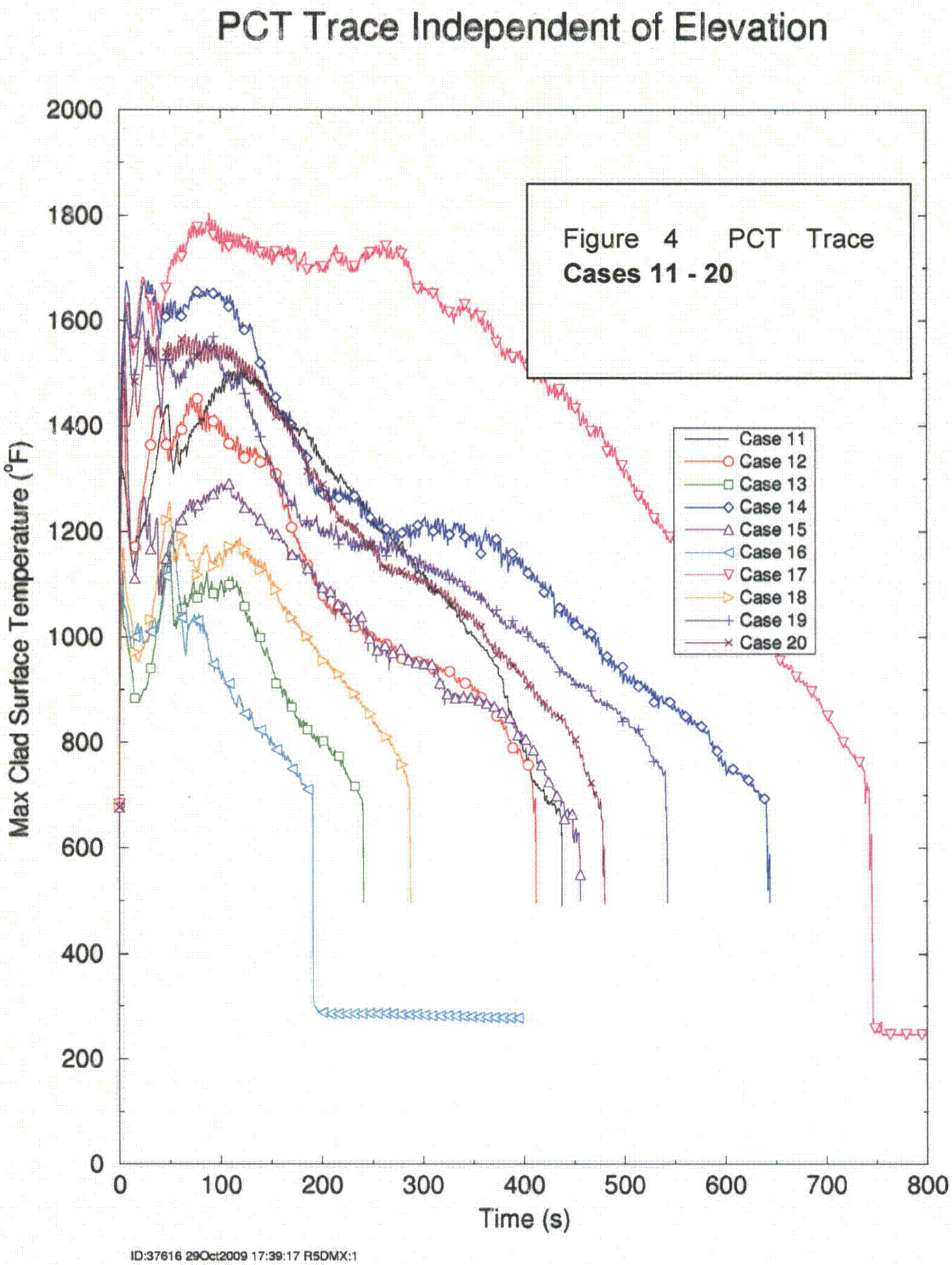

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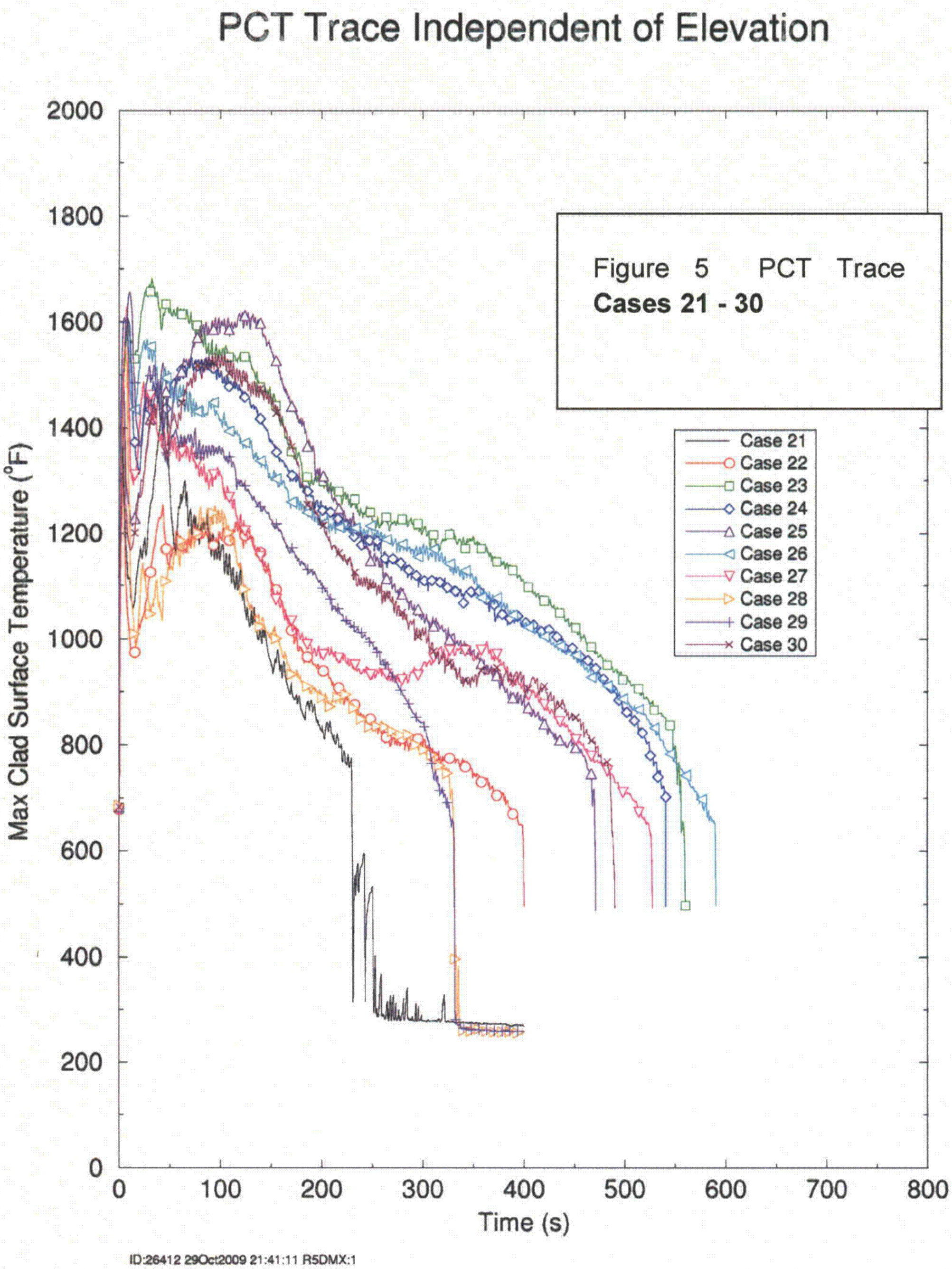

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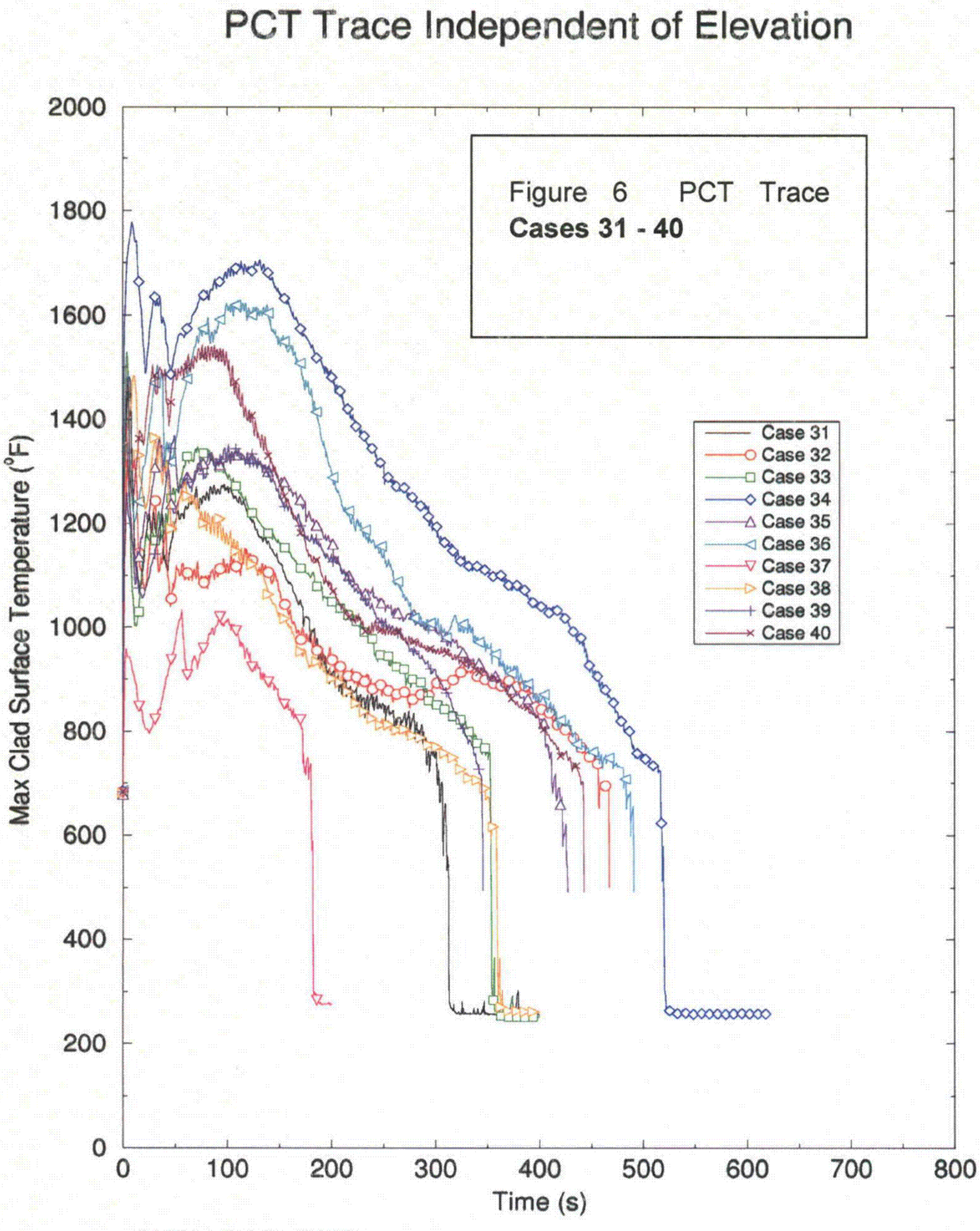

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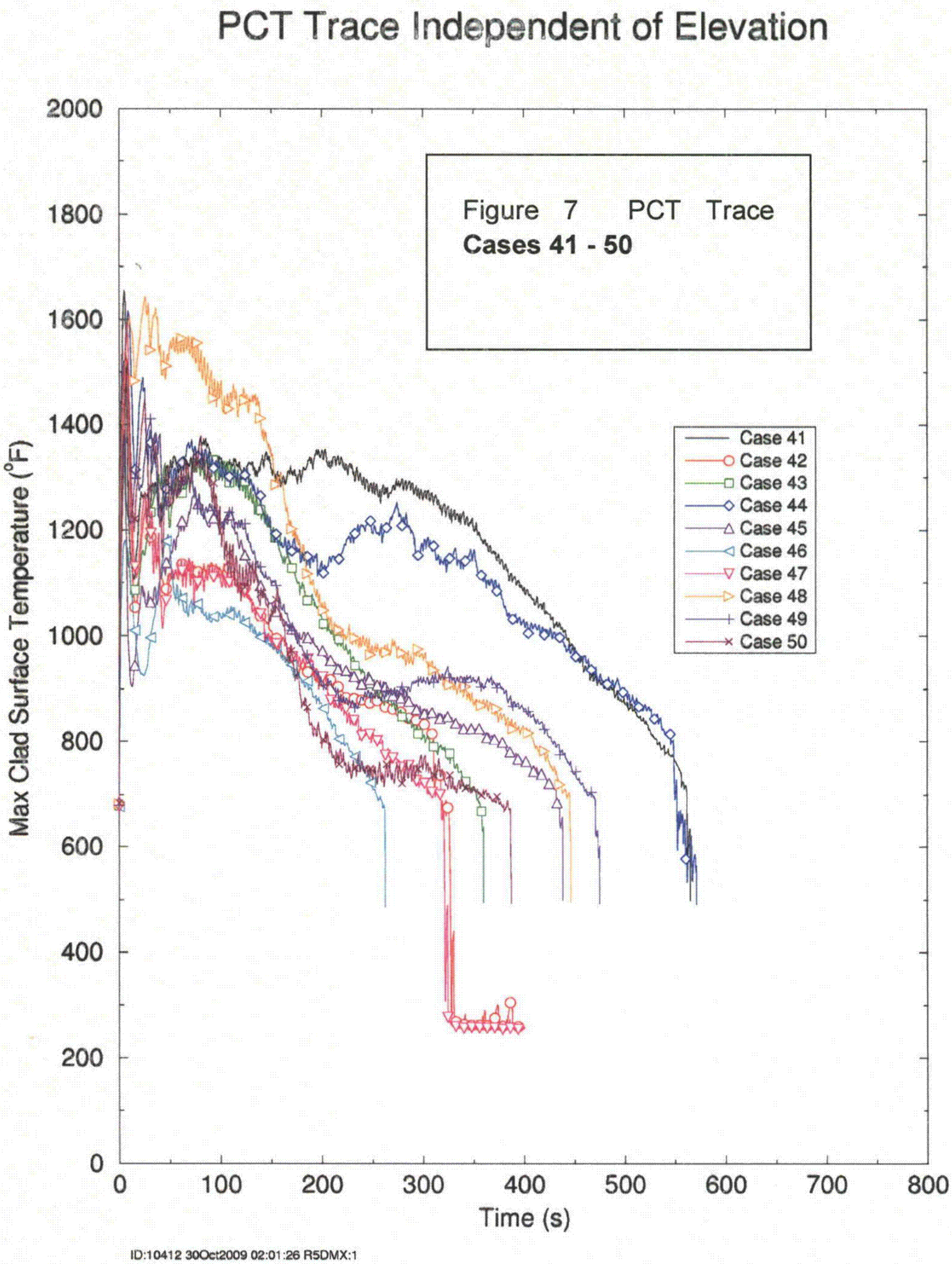

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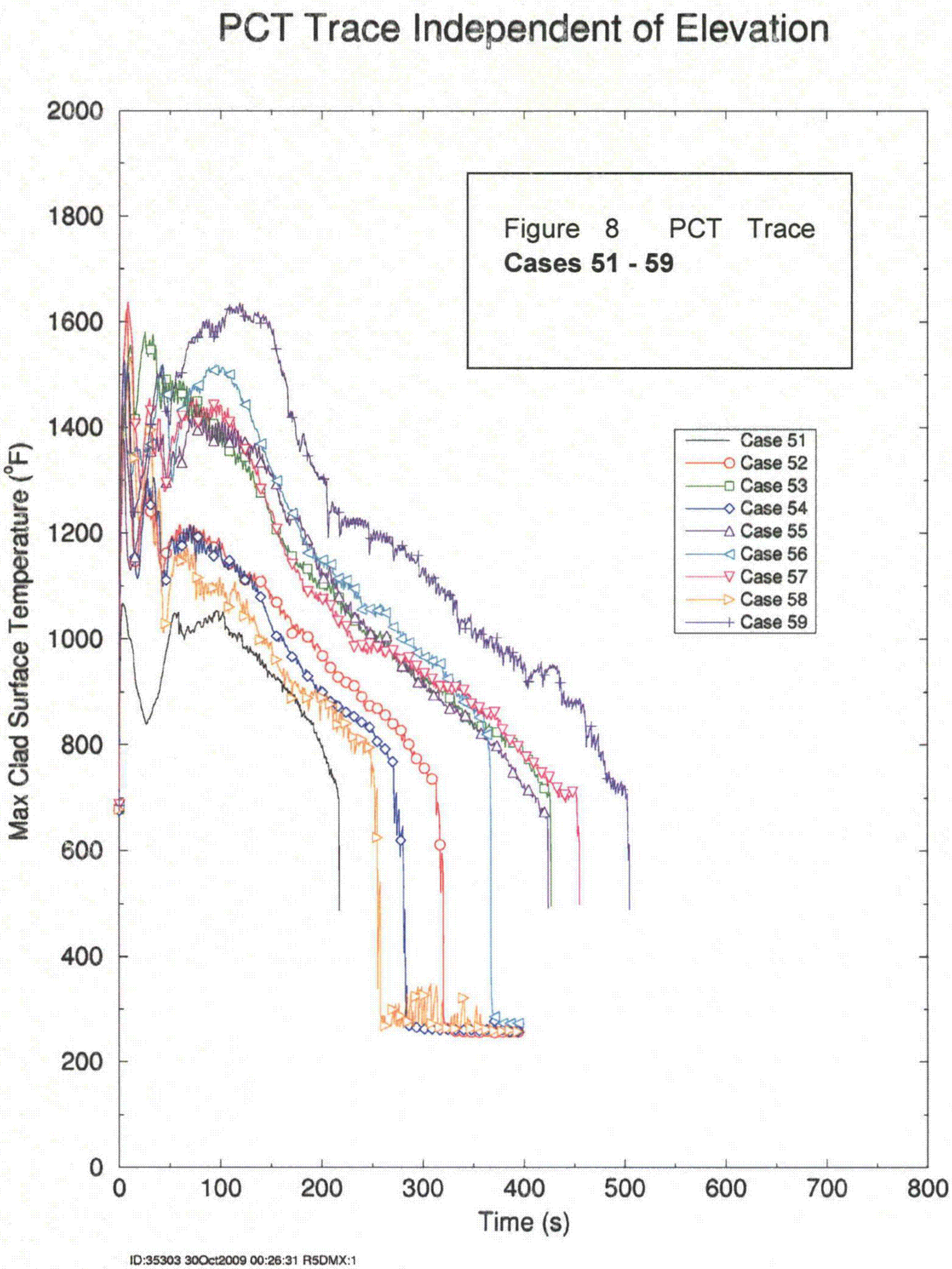

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**Request 10:**
Please provide the decay heat curve multiplier and decay power versus time applied to the limiting RLBLOCA PCT evaluation.

**Response:**
Table 2-1 in ANP-2853(P) provides the sampled decay heat multiplier for the limiting transient case as 0.9896. The decay heat curve from the limiting PCT case is shown in Figure 9.

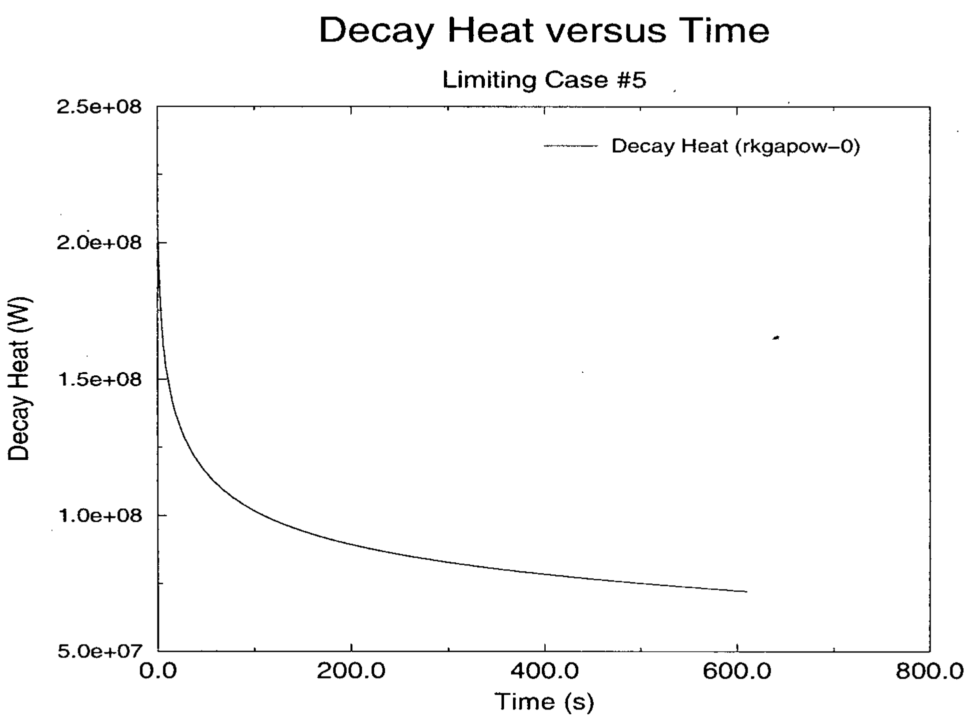


Figure 9

**References:**

1. EMF-2103(P)(A) Revision 0, "Realistic Large Break LOCA Methodology," Framatome ANP, Inc., April 2003

2. ANP-2853(P) Revision 0, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Summary Report," January 2010

3. EMF-2994(P) Revision 4, "RODEX4: Thermal-Mechanical Fuel Rod Performance Code Theory Manual," December 2009

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4. ANF-90-145(P)(A), "RODEX3 Fuel Rod Thermal-Mechanical Response Evaluation Model," April 1996

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ATTACHMENT – ANF-90-145(P)(A), Appendix B (Radial Temperature Distribution)
Pages B-1 through B-5 from the RODEX3 topical report
(5 Pages)

# APPENDIX B
# RADIAL TEMPERATURE DISTRIBUTION
# -(CYLTEM SUBROUTINE)-