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W3F1-2004-0036

June 17, 2004

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

SUBJECT: License Amendment Request NPF-38-258
To Modify Technical Specification (TS) 5.3.1, Fuel Assemblies and TS 6.9.1.11.1, Core Operating Limits Report
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCE: 1. Waterford 3 letter to the NRC dated November 13, 2003, License Amendment Request NPF-38-249 Extended Power Uprate (W3F1-2003-0074)

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests an amendment to the Waterford Steam Electric Station, Unit 3 (Waterford 3) Technical Specifications (TS).

The amendment will modify TS 5.3.1, Fuel Assemblies, to allow a limited number of lead test assemblies (LTAs) and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods. The proposed change will also include ZIRLO™ as an acceptable fuel rod cladding which is consistent with 10 CFR 50.46. To support consistency with NUREG-1432, *"Standard Technical Specifications Combustion Engineering Plants,"* some of the information in TS 5.3.1 will be relocated to TS 5.6.1. Waterford 3 submitted to the NRC, by letter dated November 13, 2003 (Reference 1), a proposed change for an extended power uprate, which in part included a change to TS 5.3.1. The change to TS 5.3.1 proposed in the power uprate will be withdrawn such that no changes to TS 5.3.1 will be included in the power uprate submittal.

Additionally a change is proposed to TS 6.9.1.11.1 to allow the use of the Westinghouse Nuclear Physics code package and to incorporate the methodology used to support ZIRLO™ cladding material.

Entergy also proposes to delete the Index from the TSs. The Index is not part of the technical content of the TSs and therefore does not need to continue to be reviewed and approved by the NRC as part of the license amendment process.

A001

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The bases for these determinations are included in Attachment 1.

The proposed change includes new commitments.

Entergy requests approval of the proposed amendment by February 28, 2005, in order to support the spring 2005 refueling outage. Once approved, the amendment shall be implemented within 60 days. Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact Dana Millar at 601-368-5445.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 17, 2004.

Sincerely,

JEV/dm/cbh

Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)
3. List of Regulatory Commitments

Enclosure 1 - Supplemental Information Describing the Use of the Westinghouse Nuclear Physics Code Package for Waterford 3

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Attachment 1

W3F1-2004-0036

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-38 for Waterford Steam Electric Station, Unit 3 (Waterford 3). The proposed change will modify Technical Specifications (TS) 5.3.1 and 5.6.1 to allow a limited number of lead test assemblies (LTAs), limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, and will allow the use of ZIRLO™ as an acceptable fuel rod cladding.

In addition, the proposed change will supplement the list of analytical methods referenced in TS 6.9.1.11 to allow the use of a Westinghouse Nuclear Physics code package and to allow the use of ZIRLO™ cladding material.

The deletion of the Index is also proposed. The Index does not include any technical information and therefore any changes to it should not require review or approval by the NRC. Its removal is purely administrative.

Approval of the proposed amendment is requested by February 28, 2005 in order to support the spring 2005 refueling outage.

2.0 PROPOSED CHANGE

The proposed change will revise Technical Specification (TS) 5.3.1, TS 5.6.1, and TS 6.9.1.11.

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The proposed change will allow the use of LTAs, ZIRLO™ fuel cladding, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods. The existing TS 5.3.1 states the following:

> The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a nominal total weight of 1830 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. Assemblies shall have a maximum nominal enrichment of 5.0 weight percent U-235 and meet the final storage requirements described in Section 5.6.

The revised wording will be consistent with NUREG-1432 and is as follows:

> The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO™ fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

The following changes are proposed to support the modified wording:

- The active fuel length of 150 inches will be deleted from the current TS wording. It is currently reflected in Table 4.2-1 in the Waterford 3 Final Safety Analysis Report (FSAR). Information contained in the FSAR is adequately controlled under the 10 CFR 50.59, *"Changes, Tests, and Experiments"* review process.
- The nominal value of 1830 grams of uranium for each fuel rod will be deleted from the current TS and added to the FSAR. Information contained in the FSAR is adequately controlled under the 10 CFR 50.59 review process.
- The initial core loading enrichment is no longer applicable and will be deleted.
- To support consistency with NUREG-1432, *"Standard Technical Specifications Combustion Engineering Plants"* the maximum enrichment of 5.0 weight percent U-235 will be moved to TS Section 5.6.1 which discusses spent fuel racks. A new item "h." will be added to TS Section 5.6.1 as follows:

  h. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

  No change is proposed to the new fuel storage racks TS as the new fuel storage racks are currently not used at Waterford 3.

Waterford 3 submitted to the NRC, by letter dated November 13, 2003, a proposed change for an extended power uprate, which in part included a change to TS 5.3.1. The change to TS 5.3.1 proposed in the power uprate submittal will be withdrawn such that no changes to TS 5.3.1 will be included in the power uprate.

In summary, the proposed change will modify the wording of TS 5.3.1, which describes the composition of the fuel assemblies that are used at Waterford 3. As a result of this change, some of the current information in TS 5.3.1 will be moved to the FSAR, a licensee controlled document. The fuel assembly maximum fuel enrichment will be relocated within the TSs to Section 5.6.1. The proposed change is consistent with NUREG-1432, using the site specific number of fuel assemblies and fuel rods in each fuel assembly. The change allows a limited number of LTAs, the use of ZIRLO™ fuel cladding, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations. The proposed change requires that the fuel designs will be analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases.

TS 6.9.1.11, Core Operating Limits Report COLR

The proposed changes to TS 6.9.1.11 include the addition of a Westinghouse Nuclear Physics code package (PHOENIX-P/ANC); the addition of a Westinghouse physics code (PARAGON); and the addition of a topical report related to ZIRLO™ fuel cladding.

Westinghouse Nuclear Physics Code Package

Entergy is transitioning from the ROCS and DIT computer codes (currently TS 6.9.1.11.1, item1) to the following Westinghouse NRC approved methodologies, which will be added as a new reference in TS 6.9.1.11.1:

- WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System For Pressurized Water Reactor Cores"
- WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code"
- WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery"

WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON," will also be added to the list of references in the COLR. This WCAP allows the PARAGON code to be used as a replacement for the PHOENIX-P lattice code. Westinghouse has not published the "-A" version, however, the NRC has approved the topical report for use. The NRC safety evaluation (SE) for WCAP-16045-P includes a condition that states: "The PARAGON code can be used as a replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies." The core reload analysis performed by Westinghouse will transition from PHOENIX-P to PARAGON and may be used in the final reload analysis for the spring 2005 refueling outage. Therefore, reference to both codes is appropriate.

A new reference that will support the use of ZIRLO™ fuel cladding (proposed for use in the Waterford 3 core design in the fall of 2006) will be added to the list of analytical methodologies. Topical report CENPD-404-P-A, "Implementation of ZIRLO™ Material Cladding in CE Nuclear Power Fuel Assembly Designs" summarizes the ZIRLO™ material properties as they pertain to fuel rod cladding and provides an evaluation of these properties and the correlations that Westinghouse intends to use in design and licensing analysis activities. In addition, CENPD-404-P-A identifies the specific CENP topical reports that would be impacted by the implementation of ZIRLO™ cladding, and describes the substitutions that would be required as a result of the proposed ZIRLO™ implementation.

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Entergy also proposes to delete the Index from the TSs. The Index is not part of the technical content of the TSs and therefore does not need to continue to be reviewed and approved by the NRC as part of the license amendment process. No further discussion is included in relationship to this administrative change.

Summary of Proposed Changes

In summary, Entergy proposes a change to modify TS 5.3.1 that will allow the use of a limited number of LTAs, ZIRLO™ fuel cladding, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods. A change to TS 5.6.1 is also proposed with the relocation of the maximum fuel enrichment from TS 5.3.1 to TS 5.6.1. In addition, a change is proposed to TS 6.9.1.11.1 to allow the use of the Westinghouse Nuclear Physics code packages (WCAP-11596-P-A, WCAP-10965-P-A, and WCAP-10965-P-A, Addendum 1, WCAP-16045-P-A) and ZIRLO™ fuel cladding by the addition of CENPD-404-P-A.

3.0 BACKGROUND

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The Waterford 3 reactor incorporates a 16 x 16 fuel assembly design which consists of 236 fuel and poison rods, five control element guide tubes, and other structural components. The fuel assembly design enables reconstitution of an irradiated fuel assembly, i.e., removal and replacement of fuel and poison rods. Waterford 3 currently used Zircaloy-4 as the fuel cladding. A detailed description of the construction and testing associated with the Waterford 3 fuel assemblies is included in the Waterford 3 FSAR Section 4.2.

The proposed change includes the option to use ZIRLO™ as a fuel cladding material. ZIRLO™ is a zirconium-based fuel rod cladding material developed by Westinghouse which the NRC previously reviewed and approved for use. 10 CFR 50.46 has also been revised to support the use of ZIRLO™. Although Waterford 3 does not intend to use ZIRLO™ in the upcoming core reload design, the addition of ZIRLO™ as an optional fuel cladding material is consistent with the standard TSs and the CFR.

TS 6.9.1.11, Core Operating Limits Report COLR

Westinghouse Nuclear Physics Code Package

Nuclear designs for reloads and the evaluation of reload safety for Waterford 3 have been performed using the CE-ABB reload methodology. To date this methodology has been executed using the tools which constitute the CE-ABB Nuclear Physics code package (DIT and ROCS as approved in CENPD-266-P-A). The basis for this change is a transition to the Westinghouse Nuclear Physics code package with the continued application of the CE-ABB reload and safety analysis methodology. The change in the tools used to execute the reload methodology is based on the integration of technologies arising from the consolidation of the former CE-ABB nuclear entities with Westinghouse Electric Company LLC.

The Westinghouse Nuclear Physics code package is based on the ANC and PHOENIX codes, which have been reviewed and approved previously by the NRC, as described in the following topical reports:

- WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System For Pressurized Water Reactor Cores"
- WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code"
- WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery"

The Westinghouse Nuclear Physics code package has been used extensively for the design of reload cores and for evaluation of reload safety for a wide range of core sizes and fuel array sizes encompassing typical designs for Westinghouse, Combustion Engineering (CE), and Framatome designs. The capabilities and functionality of the ANC/PHOENIX technology is well known by the NRC and the nuclear industry. Based on significant experience, including benchmarks on several CE type plants, the application of the Westinghouse Nuclear Physics code package is expected to provide predictions of key core parameters that are essentially the same as those obtained with the current DIT/ROCS methodology. Margins in

nuclear design, based solely on the transition in nuclear physics methods, are expected to remain essentially unchanged. The first reload cycle for application of the Westinghouse Nuclear Physics code package for Waterford 3 may be Cycle 14 (Spring 2005).

WCAP-16045-P-A, "Qualification of the Two Dimensional Transport Code PARAGON," also be added as a new item to the list of COLR references. This WCAP, as approved by the NRC, allows the PARAGON code to be used as a replacement for the PHOENIX-P lattice code. PARAGON may be used in the final reload analysis for the spring 2005 refueling outage.

ZIRLO™ Cladding

In a continuing effort to improve fuel performance, Waterford 3 may implement ZIRLO™ cladding material for new fuel assemblies beginning in 2006. The use of ZIRLO™ clad fuel rods will substantially reduce exterior corrosion and particularly spalling experienced by some current Zircaloy-4 clad fuel rods as they approach higher burnup levels and duty cycles. The proposed TS change to TS 6.9.1.11 provides a methodology reference for the use of ZIRLO™ clad fuel rods in the Waterford 3 reactor core.

Topical report CENPD-404-P-A describes the implementation of ZIRLO™ fuel rod cladding material properties and correlations in design and safety analysis methodologies for CE design reactors and fuel. Westinghouse has performed extensive evaluations which have concluded that the application of ZIRLO™ in existing CE fuel designs does not result in any undesirable changes in predicted fuel performance or safety analysis results. While modification to CE computer codes are required to implement ZIRLO™ material properties, no modifications are required to the NRC accepted ZIRLO™ properties or analysis methodologies for CE design nuclear steam supply systems and fuel designs, design performance criteria, or regulatory acceptance criteria. Waterford 3 is a CE designed reactor and is supplied with CE designed nuclear fuel.

The ZIRLO™ topical report requires the use of specific versions of the Westinghouse Emergency Core Cooling System (ECCS) performance evaluation models for CE designed reactors (i.e., CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model" and CENPD-137, "Calculative Methods for the CE Small Break LOCA Evaluation Model"). These are currently included in the Waterford 3 TS as analytical methods used to determine the core operating limits.

4.0 TECHNICAL ANALYSIS

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The change allows a limited number of LTAs, the use of ZIRLO™ fuel cladding, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations. The proposed change requires that the fuel designs will be analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases.

TS 6.9.1.11, Core Operating Limits Report COLR

Westinghouse Nuclear Physics Code Package

To support the application of the Westinghouse Nuclear Physics code package for Waterford 3, plant specific comparisons of key physics parameters for two cycles of past plant operation are provided in Enclosure 1. The comparison provided is between actual plant operating data from Cycles 11 and 12 and design calculations using the Westinghouse Physics code package PHOENIX-P/ANC. In future Waterford 3 core reload evaluations Westinghouse may use PARAGON in place of PHOENIX-P as allowed by the NRC safety evaluation for WCAP-16045-P which states: "The PARAGON code can be used as a replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in NRC-approved methodologies."

Implementation of the Westinghouse Nuclear Physics code package requires no functional changes in the current reload methods. There are no changes in the safety analyses or safety analysis methods. Changes are limited to those necessary to support effective and accurate electronic transfer of data from the Westinghouse Nuclear Physics code package to the downstream interface codes with components of the current reload methodology.

The first reload cycle for application of the Westinghouse Nuclear Physics code package for Waterford 3 may be in the 2005.

The NRC SE for the Westinghouse topical report WCAP-16045-P, Revision 0, "Qualification of the Two-dimensional Transport Code Paragon," was approved by the NRC on March 18, 2004. The SE contained two conditions which are addressed below.

Condition 1

The PARAGON code can be used as a replacement for the PHOENIX-P lattice code, wherever the PHOENIX-P code is used in the NRC-approved methodologies.

Response 1

This letter includes a request to use the PHOENIX-P code. As allowed by this Condition, Westinghouse may use PARAGON as replacement code for PHOENIX-P code when evaluating the Waterford 3 core reload. Therefore, this letter also includes a request to include WCAP-16045-P as a reference in the TS.

Condition 2

The data base is insufficient to enable the staff to reach a conclusion regarding PARAGON's ability to predict depletion characteristics for a MOX fueled core at this time.

Response 2

The Waterford 3 core is not a MOX fueled core; therefore, this condition does not apply.

ZIRLO™ Cladding

Topical report CENPD-404-P, which describes the implementation of ZIRLO™ fuel rod cladding material properties and correlations in Westinghouse design and safety analysis methodologies for Combustion Engineering designed pressurized water reactors and fuel, was approved by the NRC subject to five conditions. The five conditions and Waterford 3's response are as follows:

Condition 1

The corrosion limit as predicted by the best-estimate model will remain below 100 microns for all locations of the fuel.

Waterford 3 Response

The maximum allowable corrosion limit of 100 microns will be added to the Waterford 3 FSAR. The corrosion thickness will be calculated using the best estimate models and methods described in topical report CENPD-404-P.

Condition 2

All the conditions listed in the safety evaluations for all the CENPD methodologies used for ZIRLO™ fuel analysis will continue to be met, except that the use of ZIRLO™ cladding in addition to Zircaloy-4 cladding is now approved.

Waterford 3 Response

Waterford 3 will continue to abide by the conditions listed in the safety evaluations for all CENPD methodologies used for the analysis of ZIRLO™ fuel. This will be accomplished through the reload process that is employed.

Condition 3

All CENP methodologies will be used only within the range for which ZIRLO™ data was acceptable and for which the verifications discussed in CENPD-404-P and responses to requests for additional information were performed.

Waterford 3 Response

ZIRLO™ data ranges for the methodologies in which they are used will be verified during the reload/core design process that is employed for use of methodologies.

Condition 4

Until data is available demonstrating the performance of ZIRLO™ cladding in CENP design plants, the fuel duty will be limited for each CENP designed plant with some provision for adequate margin to account for variations in core design (e.g., cycle length, plant operating conditions, etc.). Details of this condition will be address on a plant specific basis during approval to use ZIRLO™ in a specific plant.

Waterford 3 Response

During the initial fuel cycle in which ZIRLO™ will be used, the modified fuel duty index (mFDI) will be limited until data is available demonstrating the performance of ZIRLO™ cladding at the CE lead plant.

The maximum mFDI calculated based on actual 16 x 16 CE designed fuel is approximately 590. To provide adequate margin to account for variations in core design, 110% of the approximate 590 value (652) is used for the majority of the ZIRLO™ clad fuel pins. For a fraction of the fuel pins in a limited number of assemblies (no more than eight fuel assemblies), the mFDI limit is 120% of the approximate 590 value (712). The mFDI values of 652 and 712, with the aforementioned limitations will be used as upper design limits for the Waterford 3 fuel.

If the mFDI and measured oxide thickness from the CE lead plant utilizing ZIRLO™ correlate as expected or is conservative relative to predictions, Waterford 3 will no longer restrict the mFDI except as required to meet the 100 micron oxide limit.

Condition 5

The burnup limit for this approval is 60 GWD/MTU.

Waterford 3 Response

The maximum integrated rod burnup is 60 GWD/MTU as stated in FSAR Section 4.3A3.1.3, "Maximum Fuel Rod Burnup."

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met. Entergy Operations, Inc. (Entergy) has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR.)

5.2 No Significant Hazards Consideration

The proposed change will modify several Technical Specifications (TS) to support core reload activities for Waterford Steam Electric Station, Unit 3 (Waterford 3).

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The wording of the TS 5.3.1, which describes the composition of the fuel assemblies that are used at Waterford 3 will be modified. The change allows a limited number of lead test assemblies with approved exemptions, and limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations. The proposed change requires that the fuel designs will be analyzed with

applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases.

As a result of this change, some of the current information in TS 5.3.1 will be moved to the Final Safety Analysis Report (FSAR). The relocation of information to the FSAR will improve the usability and readability of the TSs without changing any design requirements of the facility. The fuel assembly maximum fuel enrichment will be relocated within the TSs to Section 5.6.1. The proposed change is consistent with NUREG-1432, using the site specific number of fuel assemblies and fuel rods in each fuel assembly.

TS 6.9.1.11, Core Operating Limits Report COLR

A change is also proposed to analytical methodologies used to determine the core operating limits addressed by individual TS. The proposed change will add the following NRC approved Westinghouse Nuclear Physics code package to the administrative controls section of the TSs:

- WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System For Pressurized Water Reactor Cores"
- WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code"
- WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery"

WCAP-16045-P-A ("Qualification of the Two-Dimensional Transport Code PARAGON") will also be added to the list of methodologies included in the TSs. This allows the PARAGON code to be used as a replacement for the PHOENIX-P code.

TS 6.9.1.11 will also be modified to incorporate a reference to CENPD-404-P-A, "Implementation of ZIRLO™ Material Cladding in CE Nuclear Power Fuel Assembly Designs."

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And finally, an administrative change is proposed to delete the Index from the Technical Specifications.

Entergy 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. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

   The proposed change allows the use of a limited number of lead test assemblies; the use of limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods; and the use of methods required for the implementation of ZIRLO™ clad fuel rods.

Inasmuch as the revision identifies codes previously approved by the NRC for CE cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident.

The proposed change in part represents a relocation of a portion of the information previously located in the TSs design features section to the FSAR, which is controlled under 10 CFR 50.59, *"Changes, Tests, and Experiments."* This change is administrative in nature because the design requirements for the facility remain the same.

The proposed change does not remove or modify any of the design requirements for the facility or affect any accident initiators, conditions or assumption for an accident previously evaluated.

TS 6.9.1.11, Core Operating Limits Report COLR

The proposed amendment identifies a change in the nuclear physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits and includes minor editorial changes which do not alter the intent of stated requirements. The proposed change also allows the use of methods required for the implementation of ZIRLO™ clad fuel rods. Inasmuch as the proposed change identifies codes previously approved by the NRC for CE cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. Parameter limits specified in the site specific COLR are not changed from the values presently required by TSs. Future changes to the calculated values of such limits may only be made using NRC approved methodologies, must be consistent with all applicable safety analysis limits, and are controlled by the 10 CFR 50.59 process. Assumptions used for accident initiators and/or safety analysis acceptance criteria are not changed by this change.

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The proposed change is administrative in nature and does not affect any system or component functional requirements. This change does not affect the operation of the plant or affect any component that is used to mitigate the consequences of any accident.

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

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

Response: No.

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The proposed change allows the use of methods required for the implementation of ZIRLO™ clad fuel rods. Inasmuch as the revision identifies codes previously

approved by the NRC for CE cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident.

In addition, the proposed change allows the use of a limited number of lead test assemblies. The proposed change is administrative in nature. Prior to the use of lead test assemblies, fuel designs will be analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases to assure no new or different kind of accident from any accident previously evaluated will be created.

And finally the proposed change allows the relocation of a portion of the information previously located in the TSs design features section to the FSAR. This change is administrative in nature and does not create a new or different type of accident than previously evaluated because the design requirements for the facility remain the same.

The proposed change does not remove or modify any of the design requirements for the facility or affect any accident initiators, conditions or assumption for an accident previously evaluated.

TS 6.9.1.11, Core Operating Limits Report COLR

The proposed change identifies a change in the Nuclear Physics codes used to confirm the values of selected cycle-specific reactor physics parameter limits contained in the COLR. The proposed change also allows the use of methodologies required for the implementation of ZIRLO™ clad fuel rods. Neither of these changes results in a change the physical plant or the modes of operation defined in the facility license.

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The proposed change is administrative in nature and does not affect any system or component functional requirements. This change does not affect the operation of the plant or affect any component that is used to mitigate the consequences of any accident.

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

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

Response: No.

TS 5.3.1, Fuel Assemblies and TS 5.6.1, Criticality

The proposed change allows the use of methods required for the implementation of ZIRLO™ clad fuel rods. Inasmuch as the revision identifies codes previously approved by the NRC for CE cores, the amendment is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident.

In addition, the proposed change allows the use of a limited number of lead test assemblies. The proposed change is administrative in nature. Prior to the use of lead test assemblies, fuel designs will be analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to ensure compliance with any safety analysis acceptance criteria.

And finally the proposed change allows the relocation of a portion of the information previously located in the TSs design features section to the FSAR. This change is administrative in nature and does not create a new or different type of accident than previously evaluated because the design requirements for the facility remain the same.

The proposed change does not remove or modify any of the design requirements for the facility or affect any accident initiators, conditions or assumption for an accident previously evaluated.

TS 6.9.1.11, Core Operating Limits Report COLR

The individual specifications continue to require operation of the plant within the bounds of the limits specified in COLR. Benchmarking has shown that uncertainties for the Westinghouse Physics code system (ANC/PHOENIX-P) yields are essentially the same or less than those obtained for the current ROCS/DIT methodology. Future changes to the values of these limits by the licensee may only be developed using NRC approved methodologies, remaining consistent with all applicable plant safety analysis limits addressed in the Safety Analysis Report, which are controlled by the 10 CFR 50.59 process. The relocation of the supplement numbers, revision numbers, and approval dates related to the analytical methods listed in the COLR does not affect the margin of safety. The analysis will continue to be performed using NRC approved methodology. Safety analysis acceptance criteria are not being altered by this change.

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The proposed change is administrative in nature and does not affect any system or component functional requirements. Safety analysis acceptance criteria are not being altered by this change.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Entergy 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.

5.3 Environmental Considerations

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 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

Attachment 2

W3F1-2004-0036

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5.3 REACTOR CORE

INSERT 1

FUEL ASSEMBLIES

5.3.1 ~~The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 236 fuel rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a nominal total weight of 1830 grams uranium. The initial core loading shall have a maximum enrichment of 2.91 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading. Assemblies shall have a maximum nominal enrichment of 5.0 weight percent U-235 and meet the final storage requirements described in Section 5.6.~~

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 87 control element assemblies.

5.4 NOT USED

5.5 METEOROLOGICAL TOWERS LOCATION

5.5.1 The primary and backup meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with:

a. A normal k_{eff} of less that or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

b. A nominal 10.185 inch center-to-center distance between fuel assemblies placed in Region 1 (cask storage pit) spent fuel storage racks.

c. A nominal 8.692 inch center-to-center distance between fuel assemblies in the Region 2 (spent fuel pool and refueling canal) racks, except for the four southern-most racks in the spent fuel pool which have an increased N-S center-to-center nominal distance of 8.892 inches.

d. New or partially spent fuel assemblies may be allowed unrestricted storage in Region 1 racks.

e. New fuel assemblies may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern" as illustrated in Figure 5.6-1.

f. Partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure 5.6-2 may be allowed unrestricted storage in the Region 2 racks.

g. Partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 5.6-2 may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern", as illustrated in Figure 5.6-1, with spent fuel in the "acceptable range" of Figure 5.6-3.

h. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

5.6.2 The k_{eff} for new fuel stored in the new fuel storage racks shall be less than or equal to 0.95 when flooded with unborated water and shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL. When fuel is being stored in the cask storage pit and/or the refueling canal, these areas will also be maintained at +40.0 MSL.

CAPACITY

5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal. The heat load from spent fuel stored in the refueling canal racks shall not exceed 1.72x10E6 BTU/Hr. Fuel shall not be stored in the spent fuel racks in the cask storage pit or the refueling canal unless all of the racks are installed in each respective area per the design.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

CORE OPERATING LIMITS REPORT COLR (Continued)

6) "CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System," CENPD-107. (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

7) "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

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

INSERT 2

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 Not Used

Insert 1:

The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

Insert 2:

9) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

10) "Qualification of the PHOENIX-P/ANC Nuclear Design System For Pressurized Water Reactor Cores," WCAP-11596-P-A; "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A; and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery," WCAP-10965-P-A Addendum 1. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

11) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

Attachment 3

W3F1-2004-0036

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The nominal value of 1830 grams of uranium for each fuel rod will be added to the FSAR.	x		
Prior to the use of lead test assemblies (LTAs), fuel designs will be analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases and to assure no new or different kind of accident from any accident previously evaluated will be created.		Prior to incorporating LTAs in the core and while LTAs are in the core.	
The maximum allowable corrosion limit of 100 microns will be added to the Waterford 3 FSAR.		Prior to the use of ZIRLO™ cladding.	
The corrosion thickness will be calculated using the best estimate models and methods described in topical report CENPD-404-P.		Prior to the use of ZIRLO™ and each core reload that contains ZIRLO™ fuel cladding material.	
ZIRLO™ data ranges for the methodologies in which they are used will be verified during the reload/core design process that is employed for use of methodologies.		Prior to the use of ZIRLO™ and each core reload that contains ZIRLO™ fuel cladding material.	

List of Regulatory Commitments (continued)

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
The upper design limits for Waterford 3 fuel will be limited to mFDI values of 652 for the majority of the fuel assemblies and 712 for a fraction of fuel pins in a limited number of assemblies (no more than eight fuel assemblies).			Limits required until data from CE lead plant is verified acceptable, at which time Waterford 3 will no longer restrict the mFDI except as required to meet the 100 micron oxide limit.

Enclosure 1 to

W3F1-2004-0036

Supplemental Information Describing the Use of the Westinghouse Nuclear Physics Code Package for Waterford 3

Supplemental Information Describing the use of the Westinghouse Nuclear Physics Code Package for Waterford 3
April 2004

ABSTRACT

This document provides benchmarking data for Waterford 3 Steam Electric Station, Unit 3 (Waterford 3). The computer programs used are part of the Westinghouse Nuclear Physics code package (ANC/PHOENIX) and were obtained from Westinghouse Electric Company LLC. The calculations were performed by Westinghouse Electric Company LLC. The results of these calculations were compared to operating data from Waterford 3. These comparisons further demonstrate the applicability of the Westinghouse Nuclear Physics code package to perform reload design calculations for Waterford 3.

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2.3-20 Waterford Unit 3 Cycle 12 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 495.3 EFPD 36

1.0 INTRODUCTION AND CONCLUSIONS

This report provides comparisons between predictions and operating data as a further demonstration of the applicability of the Westinghouse Nuclear Physics code package to perform reload design calculations for Waterford Steam Electric Station, Unit 3 (Waterford 3).

1.1 OBJECTIVE

The objective of this report is to further demonstrate the applicability of the Westinghouse Nuclear Physics code package to perform reload design calculations for Waterford 3. To this end, extensive design calculations have been performed for Waterford 3 Cycles 11 and 12, and the results are compared to actual plant operating data.

1.2 SCOPE

Westinghouse has performed core design calculations for comparison with core follow data for Waterford 3. Comparisons between measurements and predictions for Waterford 3 Cycles 11 and 12 are presented in Section 2. All methods used to generate the results detailed in this report (computer programs, model development, and data processing) are standard licensed methods used by Westinghouse Electric Company LLC.

1.3 CONCLUSIONS

This report describes the use of the Westinghouse Nuclear Physics code package as applied to model the Waterford 3 core. Calculations were performed for Waterford 3 Cycles 11 and 12 as described in Section 2. The results from these comparisons further demonstrate the applicability of the Westinghouse Nuclear Physics code package to perform reload design calculations for Waterford 3.

2. PHYSICS MODEL VERIFICATION FOR WATERFORD 3

Core physics model verification for Waterford 3 will include comparisons between measurement and predictions for Cycles 11 and 12. Waterford 3 is currently in its 13th cycle of operation. In this section, predictions made using the Westinghouse Nuclear Physics code package are compared to zero power physics test measurements and at power operating data. As stated in Section 1, the methods employed to generate the predictions reported in this section are standard licensed methods used by Westinghouse Electric Company LLC.

Waterford 3 is a Combustion Engineering (CE) reactor with a thermal rating of 3441 megawatts. The core consists of 217 assemblies of the CE 16 x 16 design.

2.1 CYCLE DESCRIPTIONS

Waterford 3 Cycle 11 began operation in November 2000 and shutdown in March 2002 after a 485.4 Effective Full Power Days (EFPD) cycle. The Cycle 11 used Guardian Grid fuel assemblies with an active fuel length of 150 inches. Erbia was used for burnable absorbers. The core loading pattern for Cycle 11 including a description of the fresh fuel and the locations of control rods is shown in Figure 2.1-1.

Waterford Cycle 12 began operation in April 2002 and shutdown in October 2003 after a 539.97 EFPD cycle. Cycle 12 used Guardian Grid fuel assemblies with an active fuel length of 150 inches. Erbia was used for burnable absorbers. The core loading pattern for Cycle 12 including a description of the fresh fuel and the locations of control rods is shown in Figure 2.1-2.

2.2 ZERO POWER PHYSICS TESTS

After each refueling, startup physics tests are conducted to verify that the nuclear characteristics of the core are consistent with design predictions. While the reactor is maintained at hot zero power (HZP) conditions, the following physics parameters are measured:

- Critical Boron Concentrations,
- Moderator Temperature Coefficient,
- Control Rod Worth, and
- Differential boron worth.

All review criteria of measured to predicted data are based on ANSI/ANS 19.6.1.

2.2.1 CRITICAL BORON CONCENTRATION

Table 2.2-1 provides the comparisons between HZP critical boron concentrations measurements and predictions for Cycles 11 and 12. The values represent all rods out (ARO) and Reference Bank in conditions. As shown, excellent agreement is demonstrated for each case with all differences well within the ±50 ppm review criteria.

2.2.2 MODERATOR TEMPERATURE COEFFICIENT

Table 2.2-2 provides the comparisons between HZP Moderator Temperature Coefficient measurements and predictions for Cycles 11 and 12. Excellent agreement is demonstrated with all differences being well within the review criteria of ±2 pcm/°F.

2.2.3 CONTROL ROD WORTH

Table 2.2-3 provides the Control Rod Worth comparisons between measurement and prediction for Cycles 11 and 12. In all cases, the agreement is within acceptance criteria. Figures 2.2-1 and 2.2-2 show the integral rod worth comparisons for the Reference Bank. The predicted rod worth and integral worth were calculated at the exact conditions which were present during the measurement. Excellent agreement is observed between measured and predicted integral worth.

2.2.4 DIFFERENTIAL BORON WORTH

Table 2.2-4 provides the Differential Boron Worth comparisons between measurement and predictions for Cycles 11 and 12. Both the measured and predicted values are obtained using the worth of the Reference Bank in pcm divided by the change in boron concentration from ARO to Reference Bank inserted. All differences are within the review criteria of +/-15%.

2.3 POWER OPERATION

2.3.1 BORON LETDOWN CURVES

Reactor coolant system boron concentrations are measured daily at the plant. Critical boron concentrations measured at or very close to hot full power, all rods out, equilibrium xenon and samarium conditions are compared to the predicted boron letdown curves for Cycles 11 and 12 in Figures 2.3-1 and 2.3-2. The predicted curves were obtained from design depletions with the three-dimensional ANC model.

2.3.2 AXIAL POWER DISTRIBUTIONS

Measured core average axial power distributions for Beginning-of-Cycle (BOC), Middle-of-Cycle (MOC) and End-of-Cycle (EOC) obtained with the incore monitoring codes using incore detector "snapshots" were compared to predicted axial distributions in Figures 2.3-3 through 2.3-8. The predicted distributions were obtained from three-dimensional ANC calculations performed for core conditions similar to those at the time of the "snapshots." Overall, the comparisons show excellent agreement between measured and predicted axial power distributions.

2.4 SUMMARY

In this report, predictions made using the Westinghouse Nuclear Physics code package are compared to zero power physics test measurements and at power operating data from Waterford 3, Cycles 11 and 12. In all cases, the predictions agree well with the measurements and produce results that are essentially the same as the current DIT/ROCS system. The agreement between the predictions and the measurements reported here further demonstrates the applicability of the Westinghouse Nuclear Physics code package to perform reload design calculations for Waterford 3.

TABLE 2.2-1

Waterford Unit 3 Cycles 11 and 12 HZP Critical Boron Concentration Comparison Between Measurement and Prediction

CYCLE	BORON CONCENTRATION (PPM)			
	CONFIGURATION	MEASURED (M)	PREDICTED (P)	DIFF. (M-P)
11	ARO	1995	2014	-19
12	ARO	2016	2031	-15

Review Criteria is ± 50 ppm

TABLE 2.2-2

Waterford Unit 3 Cycles 11 and 12 HZP Moderator Temperature Coefficient

Comparison Between Measurement and Prediction

CYCLE	MODERATOR TEMPERATURE COEFFICIENT (PCM/°F)			
	CONFIGURATION	MEASURED (M)	PREDICTED (P)	DIFF. (M-P)
11	ARO	+0.88	+1.63	-0.75
12	ARO	+0.36	+1.22	-0.86

Review Criteria is ± 2 pcm/°F

TABLE 2.2-3

Waterford Unit 3 Cycles 11 and 12 Control Rod Worth Comparison Between Measurement and Prediction

CYCLE	CONTROL ROD WORTH (PCM)			
	CEA GROUP	MEASURED (M)	PREDICTED (P)	% DIFFERENCE (M/P – 1) *100
11	B[1]	1834	1838	-0.22
	1&2	745	729	2.19
	3&6	936	887	5.52
	4&5	1288	1224	5.23
	A	1331	1269	4.89
	TOTAL[2]	6114	5948	2.79
12	B[1]	1732	1687	2.67
	2&3	932	922	1.08
	1&6&P	1002	981	2.14
	4&5	1102	1078	2.23
	A	1632	1527	6.88
	TOTAL[2]	6400	6196	3.29

Acceptance Criteria is ±15% or 100 pcm which ever is greater

[1] Reference Bank - Acceptance Criteria is ±10%

[2] Sum of all measured banks within ±10%

FIGURE 2.1-1

Waterford Unit 3 Cycle 11 Core Loading Pattern

	L	N	R	S	T	V	W	X	Y
11	P0 2	S3	S3 1	T4	S4 6	T5	S2 4	T0	R1 A**
13	S3	S2 A*	T3	S4 B	T5	S2 B	T2	S2 2	R1
15	S3 1	T3	S4 4	T5	S4 P	S1	T2 3	R3	
16	T4	S4 B	T5	S4 A	T4	S2 A	T1	R1	
17	S4 6	T5	S4 P	T4	S2 5	T1	S4 B	R3	
18	T5	S2 B	S1	S2 A	T1	S4 1	R1		
19	S2 4	T2	T2 3	T1	S4 B	R1			
20	T0	S2 2	R3	R3	R3				
21	R1 A**	R1							

Legend

Batch ID CEA Group ID

* center-most bank A CEAs are not four-fold symmetric, only two-fold.
** 4-finger mini-dual CEAs are centered on major axes of core.

FIGURE 2.1-2

Waterford Unit 3 Cycle 12 Core Loading Pattern

	L	N	R	S	T	V	W	X	Y
11	S4 2	T2	T4 1	U3	T5 6	U3	T5 4	U1	S2 **
13	T2	T0 A*	U3	T3 B	U3	T5 B	U4	U1 2	S4
15	T4 1	U3	T4 4	U4	T2 A	U4	U3 3	S2	
16	U3	T3 B	U4	T4 P	U4	T1 A	U2	S2	
17	T5 6	U3	T2 A	U4	T2 5	U2	T1 B	S2	
18	U3	T5 B	U4	T1 A	U2	T5 1	S2		
19	T5 4	U4	U3 3	U2	T1 B	S2			
20	U1	U1 2	S2	S4	S3				
21		S2 **	S4						

Legend
Batch ID
CEA
Group ID

*** center-most bank A CEAs are not four-fold symmetric, only two-fold.**
**** 4-finger mini-dual CEAs have been removed.**

FIGURE 2.2-1

Waterford Unit 3 Cycle 11 Measured versus Predicted Reference Bank Integral Rod Worth

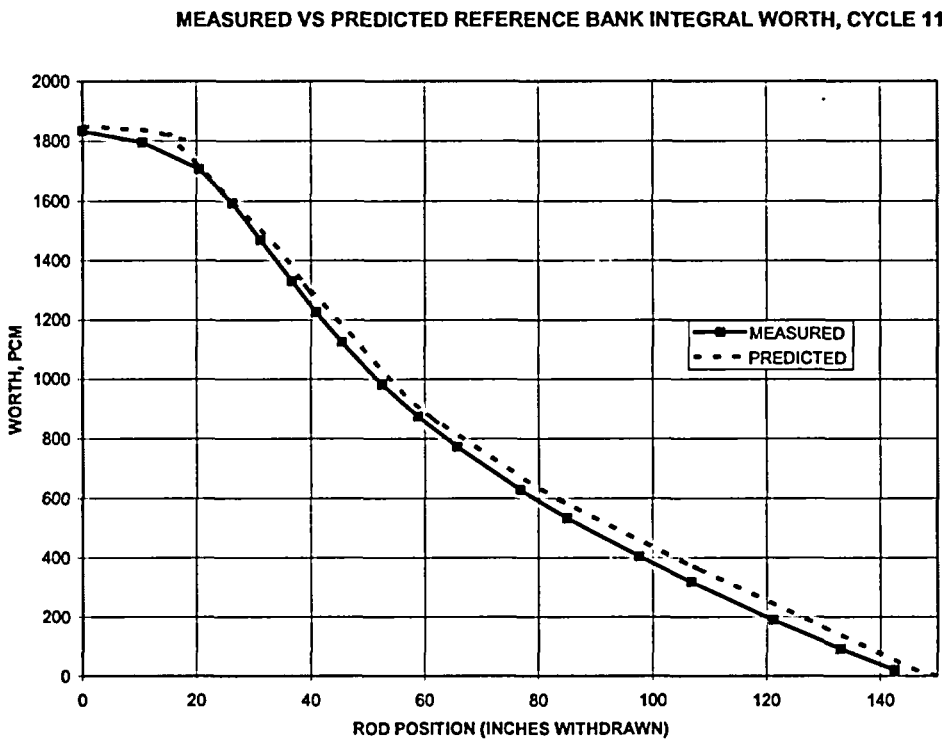

FIGURE 2.2-2

Waterford Unit 3 Cycle 12 Measured versus Predicted Reference Bank Integral Rod Worth

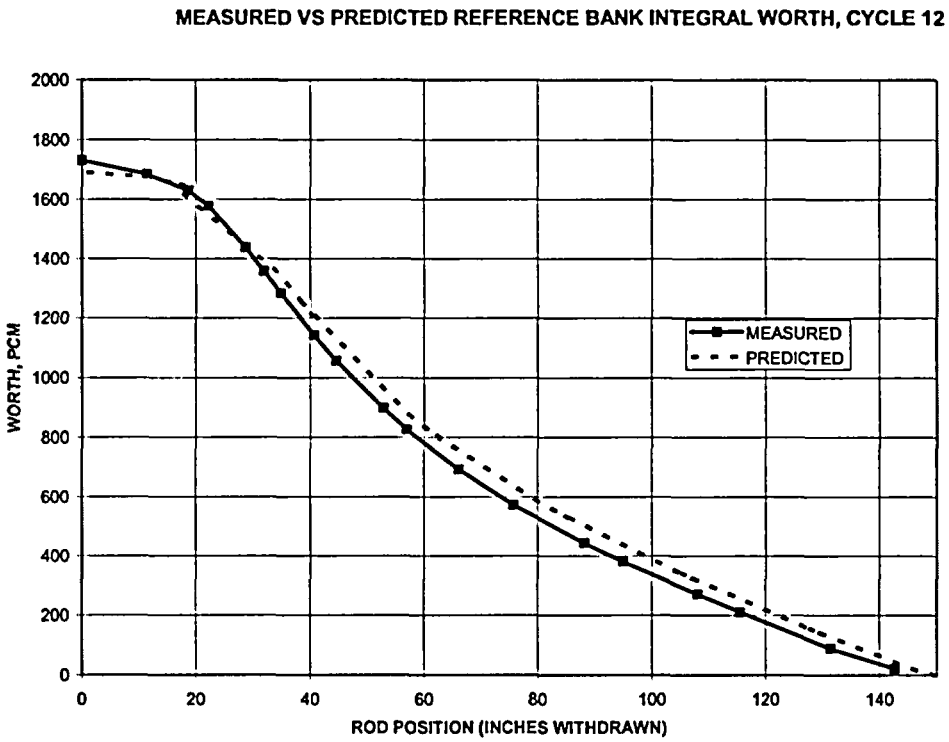

FIGURE 2.3-1

Waterford Unit 3 Cycle 11 Boron Letdown Comparison Between Measurement and Prediction

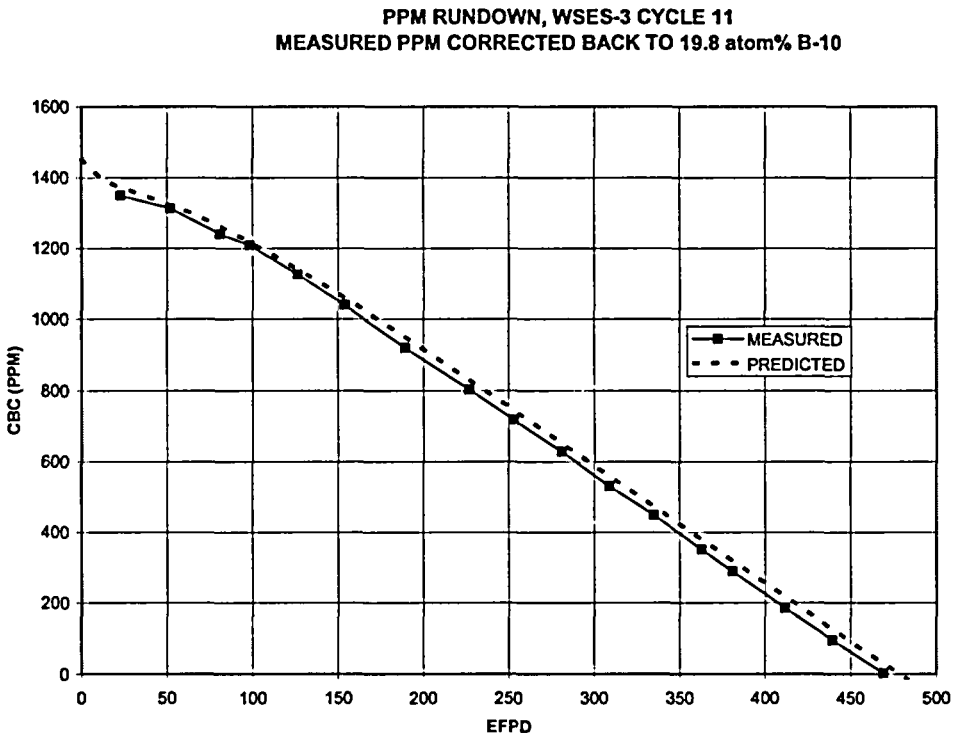

FIGURE 2.3-2

Waterford Unit 3 Cycle 12 Boron Letdown Comparison Between Measurement and Prediction

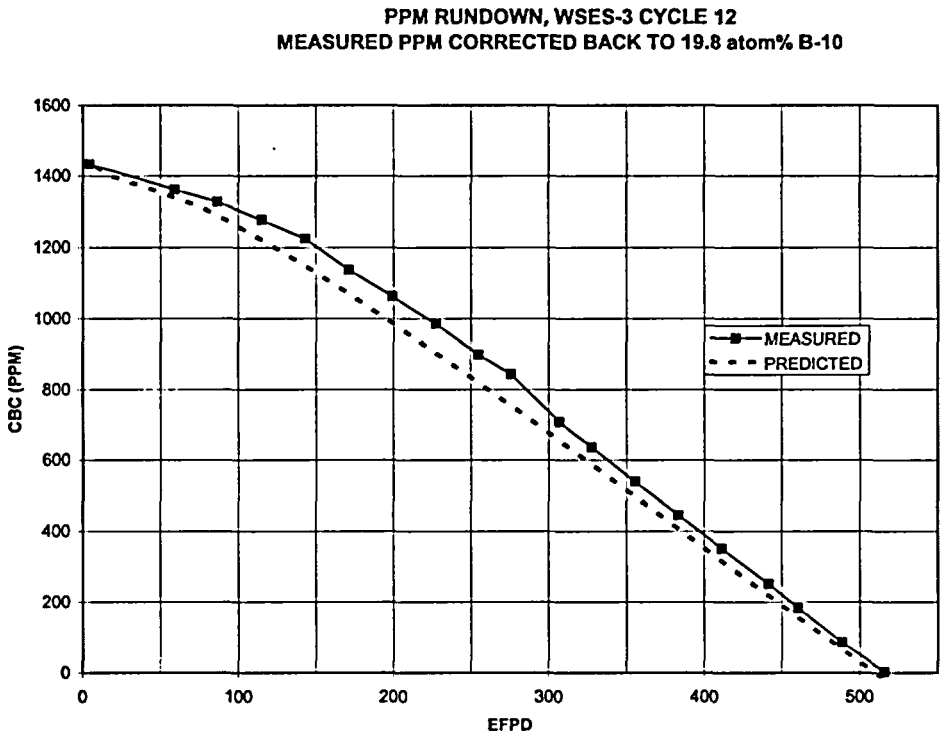

FIGURE 2.3-3

Waterford Unit 3 Cycle 11 Axial Power Distribution Comparison Between

Plant measurement and ANC – 40.2 EFPD

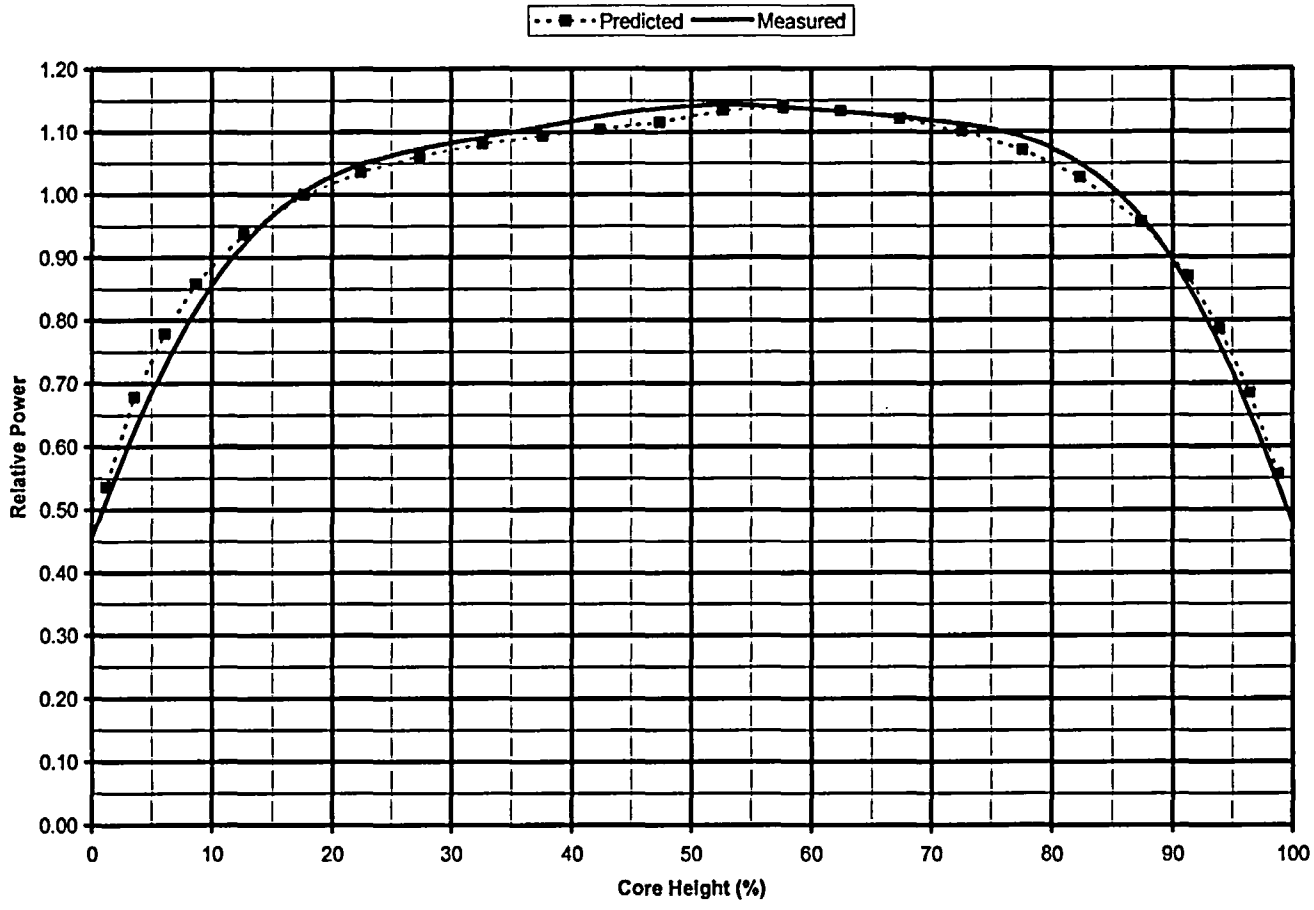

FIGURE 2.3-4

Waterford Unit 3 Cycle 11 Axial Power Distribution Comparison Between

Plant measurement and ANC – 247.0 EFPD

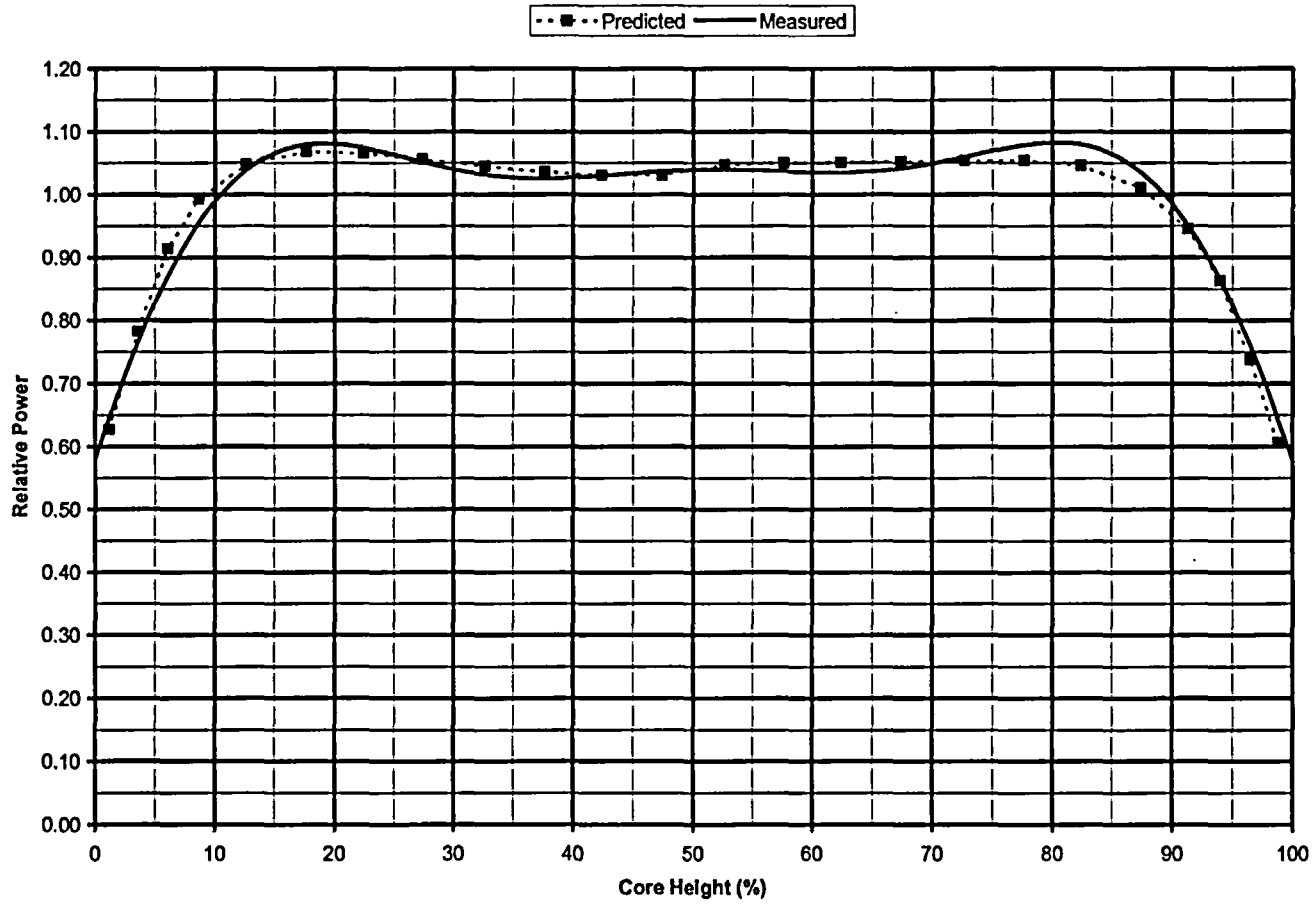

FIGURE 2.3-5

Waterford Unit 3 Cycle 11 Axial Power Distribution Comparison Between

Plant measurement and ANC – 465.0 EFPD

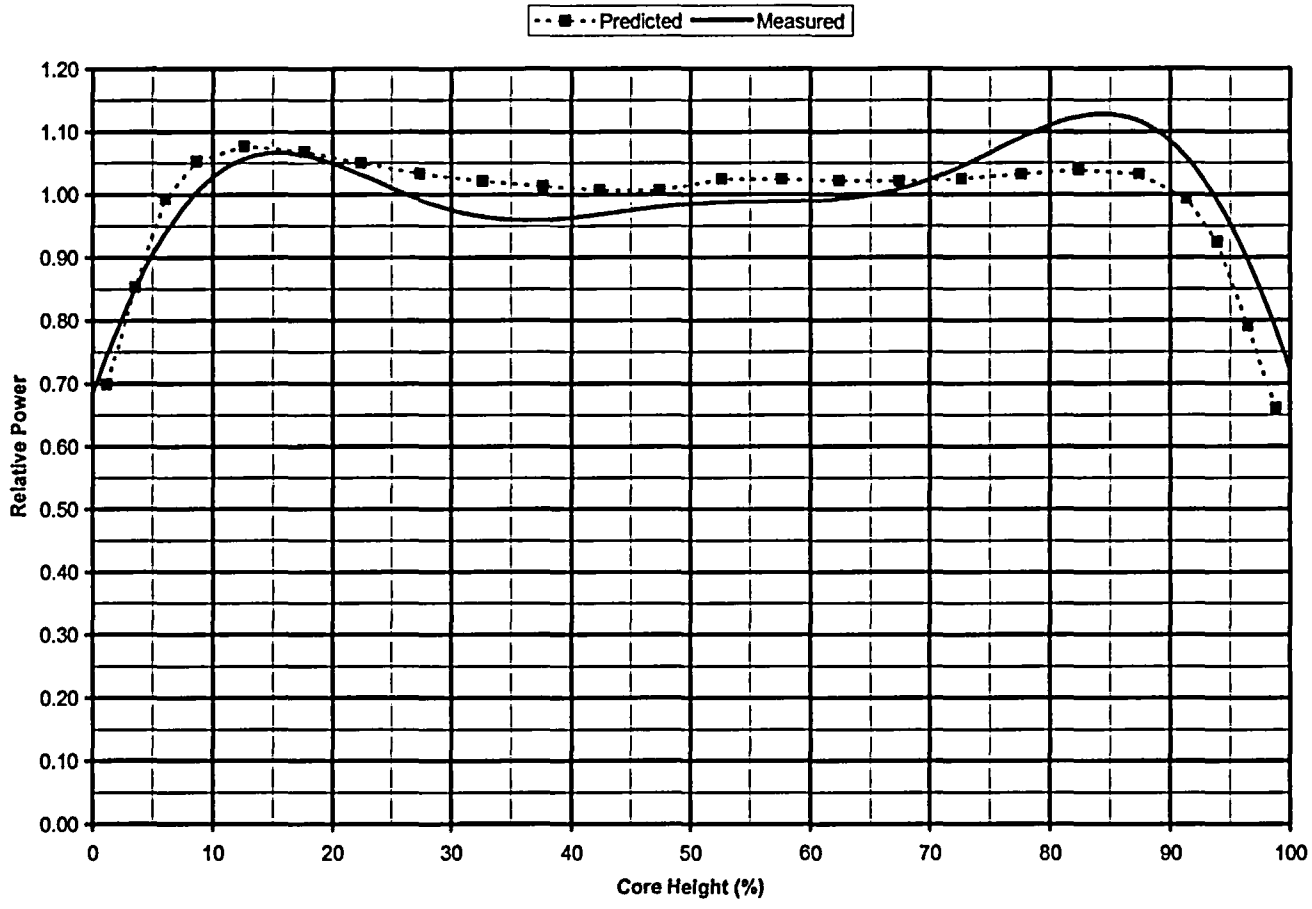

FIGURE 2.3-6

Waterford Unit 3 Cycle 12 Axial Power Distribution Comparison Between

Plant measurement and ANC – 41.3 EFPD

FIGURE 2.3-7

Waterford Unit 3 Cycle 12 Axial Power Distribution Comparison Between

Plant measurement and ANC – 258.1 EFPD

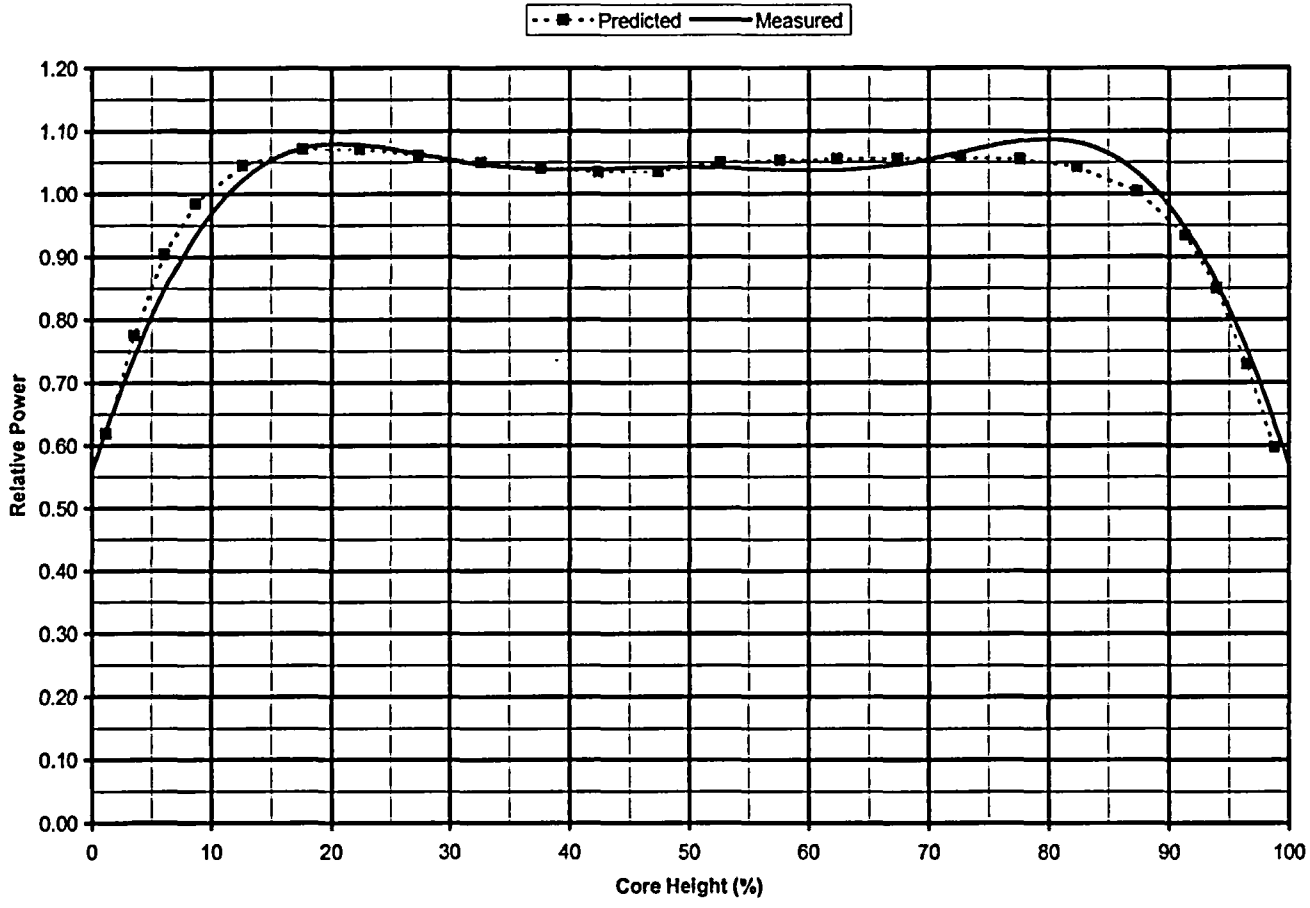

FIGURE 2.3-8

Waterford Unit 3 Cycle 12 Axial Power Distribution Comparison Between
Plant measurement and ANC – 495.3 EFPD

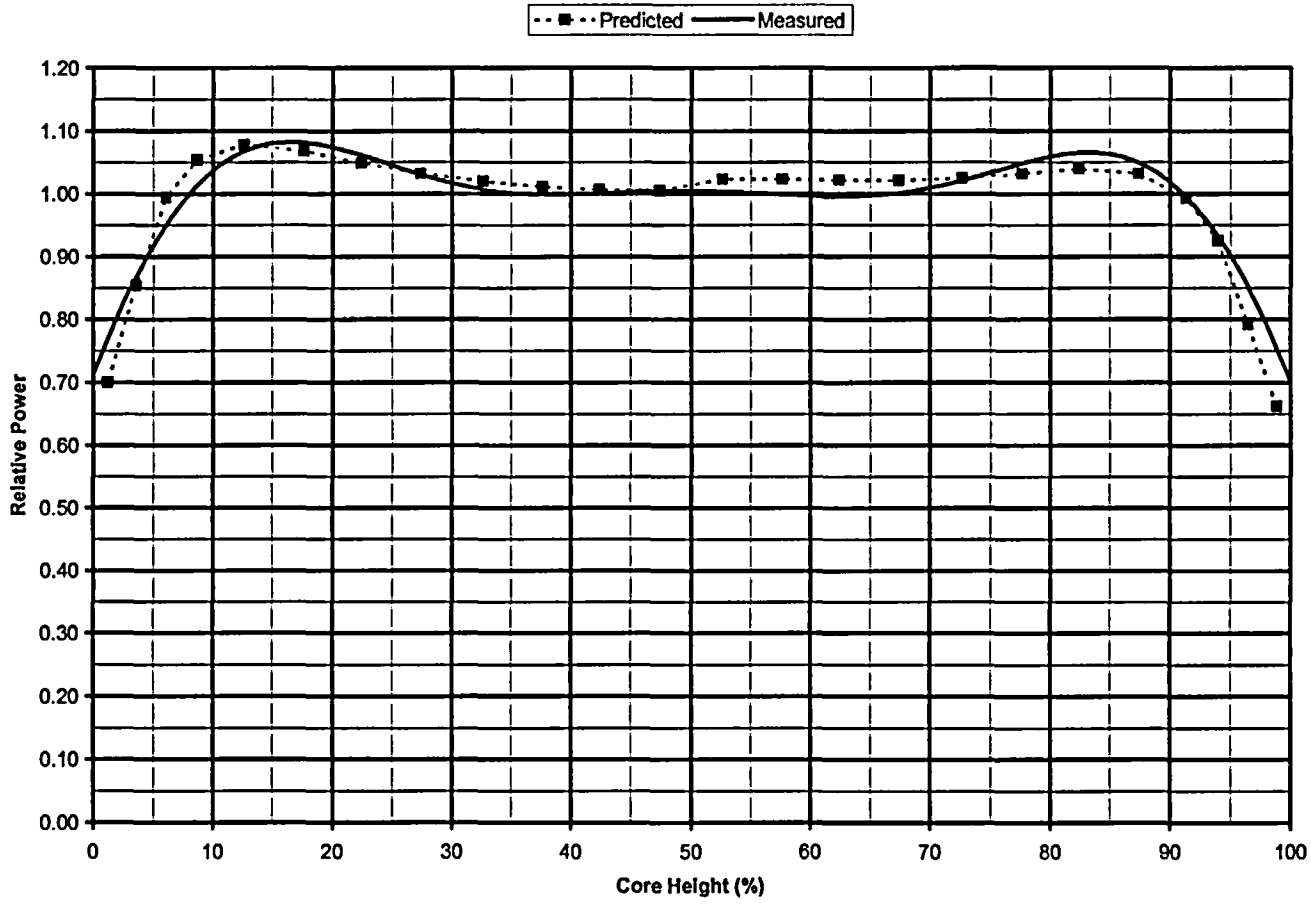

FIGURE 2.3-9

Waterford Unit 3 Cycle 11 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 40.2 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.822 0.825 -0.364	1.092 1.091 0.092	1.242 1.236 0.485	1.412 1.413 -0.071	1.280 1.280 0.000	1.385 1.398 -0.930	1.239 1.255 -1.275	1.043 1.052 -0.856	0.272 0.267 1.873
1.092 1.105 -1.176	1.228 1.236 -0.647	1.419 1.410 0.638	1.281 1.271 0.787	1.390 1.397 -0.501	1.285 1.297 -0.925	1.272 1.283 -0.857	0.754 0.753 0.133	0.180 0.172 4.651
1.242 1.249 -0.560	1.417 1.417 0.000	1.294 1.285 0.700	1.398 1.399 -0.071	1.259 1.263 -0.317	1.304 1.330 -1.955	1.170 1.184 -1.182	0.464 0.450 3.111	
1.412 1.416 -0.282	1.280 1.270 0.787	1.398 1.393 0.359	1.285 1.282 0.234	1.396 1.415 -1.343	1.223 1.248 -2.003	1.044 1.065 -1.972	0.336 0.328 2.439	
1.280 1.270 0.787	1.391 1.385 0.433	1.260 1.251 0.719	1.399 1.407 -0.569	1.298 1.321 -1.741	1.206 1.224 -1.471	0.644 0.643 0.156	0.187 0.181 3.315	
1.385 1.371 1.021	1.289 1.280 0.703	1.308 1.305 0.230	1.228 1.228 0.000	1.209 1.220 -0.902	0.700 0.693 1.010	0.262 0.248 5.645		
1.239 1.225 1.143	1.282 1.268 1.104	1.175 1.158 1.468	1.048 1.035 1.256	0.646 0.634 1.893	0.265 0.253 4.743			
1.043 1.024 1.855	0.758 0.739 2.571	0.466 0.434 7.373	0.337 0.315 6.984	0.188 0.174 8.046				
	0.273 0.258 5.814	0.181 0.172 5.233						

FIGURE 2.3-10

Waterford Unit 3 Cycle 11 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 247.0 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.811 0.805 0.745	1.053 1.043 0.959	1.196 1.188 0.673	1.436 1.436 0.000	1.256 1.252 0.319	1.419 1.422 -0.211	1.207 1.205 0.166	1.033 1.030 0.291	0.302 0.299 1.003
1.053 1.046 0.669	1.172 1.170 0.171	1.415 1.411 0.283	1.247 1.241 0.483	1.423 1.423 0.000	1.244 1.247 -0.241	1.293 1.291 0.155	0.785 0.777 1.030	0.207 0.205 0.976
1.196 1.189 0.589	1.412 1.409 0.213	1.255 1.249 0.480	1.432 1.435 -0.209	1.220 1.219 0.082	1.249 1.263 -1.108	1.203 1.210 -0.579	0.506 0.492 2.846	
1.436 1.430 0.420	1.247 1.239 0.646	1.432 1.433 -0.070	1.250 1.249 0.080	1.402 1.414 -0.849	1.172 1.188 -1.347	1.065 1.089 -2.204	0.373 0.370 0.811	
1.256 1.248 0.641	1.423 1.421 0.141	1.220 1.213 0.577	1.403 1.411 -0.567	1.234 1.249 -1.201	1.183 1.196 -1.087	0.675 0.672 0.446	0.214 0.213 0.469	
1.419 1.415 0.283	1.246 1.245 0.080	1.251 1.248 0.240	1.174 1.176 -0.170	1.185 1.201 -1.332	0.718 0.712 0.843	0.288 0.283 1.767		
1.207 1.204 0.249	1.299 1.296 0.231	1.206 1.195 0.921	1.067 1.056 1.042	0.675 0.665 1.504	0.291 0.285 2.105			
1.033 1.028 0.486	0.787 0.776 1.418	0.507 0.486 4.321	0.373 0.355 5.070	0.214 0.208 2.885				
	0.302 0.294 2.721	0.208 0.207 0.483						

FIGURE 2.3-11

Waterford Unit 3 Cycle 11 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 465.0 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.828 0.821 0.853	1.033 1.014 1.874	1.134 1.113 1.887	1.360 1.334 1.949	1.187 1.167 1.714	1.377 1.357 1.474	1.198 1.184 1.182	1.085 1.084 0.092	0.366 0.378 -3.175
1.033 1.011 2.176	1.125 1.106 1.718	1.340 1.314 1.979	1.172 1.151 1.825	1.362 1.333 2.176	1.200 1.188 1.010	1.304 1.287 1.321	0.859 0.855 0.468	0.259 0.282 -8.156
1.134 1.109 2.254	1.339 1.310 2.214	1.177 1.155 1.905	1.363 1.341 1.641	1.165 1.148 1.481	1.219 1.217 0.164	1.241 1.246 -0.401	0.580 0.579 0.173	
1.360 1.325 2.642	1.172 1.148 2.091	1.362 1.334 2.099	1.186 1.170 1.368	1.361 1.344 1.265	1.162 1.171 -0.769	1.122 1.163 -3.525	0.440 0.457 -3.720	
1.187 1.164 1.976	1.362 1.339 1.718	1.165 1.148 1.481	1.362 1.351 0.814	1.207 1.209 -0.165	1.207 1.219 -0.984	0.754 0.765 -1.438	0.266 0.281 -5.338	
1.377 1.345 2.379	1.202 1.190 1.008	1.220 1.216 0.329	1.163 1.163 0.000	1.208 1.226 -1.468	0.788 0.792 -0.505	0.343 0.369 -7.046		
1.198 1.187 0.927	1.308 1.298 0.770	1.242 1.234 0.648	1.124 1.121 0.268	0.755 0.754 0.133	0.347 0.359 -3.343			
1.085 1.083 0.185	0.860 0.853 0.821	0.580 0.574 1.045	0.442 0.436 1.376	0.266 0.281 -5.338				
	0.273 0.258 5.814	0.181 0.172 5.233						

FIGURE 2.3-12

Waterford Unit 3 Cycle 12 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 41.3 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.813 0.809 0.494	0.959 0.957 0.209	1.069 1.061 0.754	1.375 1.378 -0.218	1.154 1.146 0.698	1.382 1.379 0.218	1.116 1.095 1.918	1.218 1.203 1.247	0.367 0.353 3.966
0.959 0.964 -0.519	1.104 1.112 -0.719	1.343 1.339 0.299	1.153 1.140 1.140	1.406 1.396 0.716	1.146 1.131 1.326	1.333 1.315 1.369	1.105 1.088 1.562	0.239 0.226 5.752
1.069 1.068 0.094	1.339 1.346 -0.520	1.141 1.130 0.973	1.390 1.391 -0.072	1.208 1.202 0.499	1.392 1.389 0.216	1.247 1.242 0.403	0.580 0.566 2.473	
1.375 1.366 0.659	1.152 1.139 1.141	1.389 1.380 0.652	1.152 1.144 0.699	1.375 1.384 -0.650	1.147 1.152 -0.434	1.130 1.152 -1.910	0.380 0.374 1.604	
1.154 1.146 0.698	1.406 1.410 -0.284	1.208 1.207 0.083	1.374 1.381 -0.507	1.114 1.117 -0.269	1.208 1.224 -1.307	0.663 0.662 0.151	0.224 0.221 1.357	
1.382 1.384 -0.145	1.147 1.142 0.438	1.394 1.400 -0.429	1.147 1.150 -0.261	1.206 1.223 -1.390	0.637 0.631 0.951	0.265 0.251 5.578		
1.116 1.110 0.541	1.336 1.335 0.075	1.252 1.253 -0.080	1.135 1.144 -0.787	0.662 0.659 0.455	0.264 0.262 0.763			
1.218 1.230 -0.976	1.107 1.110 -0.270	0.586 0.578 1.384	0.398 0.391 1.790	0.216 0.205 5.366				
	0.368 0.360 2.222	0.241 0.234 2.991						

FIGURE 2.3-13

Waterford Unit 3 Cycle 12 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 258.1 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.791 0.777 1.802	0.935 0.912 2.522	1.045 1.022 2.250	1.383 1.383 0.000	1.125 1.118 0.626	1.399 1.405 -0.427	1.091 1.077 1.300	1.176 1.172 0.341	0.384 0.376 2.128
0.935 0.917 1.963	1.061 1.050 1.048	1.349 1.341 0.597	1.126 1.108 1.625	1.415 1.417 -0.141	1.130 1.122 0.713	1.367 1.366 0.073	1.097 1.092 0.458	0.259 0.250 3.600
1.045 1.025 1.951	1.346 1.340 0.448	1.118 1.099 1.729	1.408 1.408 0.000	1.178 1.173 0.426	1.418 1.428 -0.700	1.296 1.304 -0.613	0.608 0.604 0.662	
1.383 1.376 0.509	1.126 1.107 1.716	1.408 1.404 0.285	1.130 1.117 1.164	1.383 1.390 -0.504	1.101 1.110 -0.811	1.129 1.151 -1.911	0.409 0.407 0.491	
1.125 1.114 0.987	1.415 1.420 -0.352	1.178 1.175 0.255	1.382 1.389 -0.504	1.070 1.064 0.564	1.168 1.175 -0.596	0.675 0.672 0.446	0.247 0.246 0.407	
1.399 1.406 -0.498	1.131 1.126 0.444	1.418 1.436 -1.253	1.099 1.108 -0.812	1.166 1.173 -0.597	0.647 0.629 2.862	0.284 0.270 5.185		
1.091 1.084 0.646	1.368 1.377 -0.654	1.298 1.310 -0.916	1.131 1.141 -0.876	0.673 0.664 1.355	0.284 0.283 0.353			
1.176 1.185 -0.759	1.098 1.104 -0.543	0.612 0.613 -0.163	0.425 0.423 0.473	0.237 0.226 4.867				
	0.384 0.380 1.053	0.260 0.258 0.775						

FIGURE 2.3-14

Waterford Unit 3 Cycle 12 Assembly Average Power Fraction Distribution Comparison Between Plant Measurement and ANC – 495.3 EFPD

Legend: ANC Predicted Assembly Average Power Fraction
Measured Data
% Difference

0.842 0.829 1.568	0.969 0.938 3.305	1.040 1.012 2.767	1.335 1.332 0.225	1.077 1.071 0.560	1.344 1.350 -0.444	1.067 1.058 0.851	1.173 1.172 0.085	0.436 0.433 0.693
0.969 0.942 2.866	1.065 1.048 1.622	1.322 1.315 0.532	1.088 1.069 1.777	1.350 1.353 -0.222	1.089 1.085 0.369	1.338 1.348 -0.742	1.114 1.112 0.180	0.305 0.300 1.667
1.040 1.014 2.564	1.320 1.311 0.686	1.086 1.066 1.876	1.351 1.350 0.074	1.127 1.123 0.356	1.369 1.381 -0.869	1.292 1.306 -1.072	0.660 0.662 -0.302	
1.335 1.326 0.679	1.088 1.067 1.968	1.351 1.348 0.223	1.094 1.079 1.390	1.345 1.352 -0.518	1.082 1.095 -1.187	1.158 1.179 -1.781	0.471 0.473 -0.423	
1.077 1.066 1.032	1.349 1.352 -0.222	1.127 1.123 0.356	1.344 1.348 -0.297	1.064 1.055 0.853	1.193 1.194 -0.084	0.753 0.743 1.346	0.305 0.308 -0.974	
1.344 1.351 -0.518	1.089 1.086 0.276	1.369 1.388 -1.369	1.081 1.092 -1.007	1.192 1.195 -0.251	0.727 0.698 4.155	0.347 0.334 3.892		
1.067 1.058 0.851	1.339 1.346 -0.520	1.294 1.307 -0.995	1.159 1.165 -0.515	0.750 0.735 2.041	0.346 0.351 -1.425			
1.173 1.174 -0.085	1.115 1.117 -0.179	0.664 0.673 -1.337	0.490 0.489 0.204	0.292 0.281 3.915				
	0.437 0.433 0.924	0.306 0.310 -1.290						

FIGURE 2.3-15

Waterford Unit 3 Cycle 11 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 40.2 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.869 0.889 -2.250	1.220 1.210 0.826	1.359 1.327 2.411	1.481 1.476 0.339	1.359 1.336 1.722	1.459 1.475 -1.085	1.397 1.398 -0.072	1.307 1.308 -0.076	0.520 0.506 2.767
1.220 1.224 -0.327	1.438 1.415 1.625	1.499 1.483 1.079	1.362 1.329 2.483	1.463 1.472 -0.611	1.362 1.357 0.368	1.440 1.461 -1.437	1.112 1.136 -2.113	0.394 0.377 4.509
1.359 1.341 1.342	1.498 1.492 0.402	1.381 1.357 1.769	1.473 1.475 -0.136	1.344 1.324 1.511	1.405 1.403 0.143	1.427 1.452 -1.722	0.820 0.767 6.910	
1.481 1.479 0.135	1.361 1.327 2.562	1.472 1.467 0.341	1.370 1.350 1.481	1.479 1.493 -0.938	1.382 1.393 -0.790	1.344 1.377 -2.397	0.637 0.625 1.920	
1.359 1.325 2.566	1.464 1.459 0.343	1.345 1.310 2.672	1.481 1.483 -0.135	1.405 1.416 -0.777	1.465 1.476 -0.745	1.032 1.059 -2.550	0.425 0.417 1.918	
1.459 1.446 0.899	1.369 1.340 2.164	1.410 1.375 2.545	1.388 1.371 1.240	1.473 1.476 -0.203	1.084 1.083 0.092	0.575 0.548 4.927		
1.397 1.364 2.419	1.451 1.444 0.485	1.434 1.421 0.915	1.350 1.339 0.822	1.035 1.039 -0.385	0.539 0.533 1.126			
1.307 1.273 2.671	1.123 1.129 -0.531	0.825 0.734 12.398	0.668 0.613 8.972	0.426 0.403 5.707				
	0.521 0.495 5.253	0.369 0.357 3.361						

FIGURE 2.3-16

Waterford Unit 3 Cycle 11 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 247.0 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.856 0.863 -0.811	1.173 1.153 1.735	1.301 1.271 2.360	1.507 1.487 1.345	1.314 1.288 2.019	1.494 1.484 0.674	1.338 1.328 0.753	1.262 1.255 0.558	0.534 0.530 0.755
1.173 1.156 1.471	1.353 1.338 1.121	1.488 1.475 0.881	1.309 1.281 2.186	1.497 1.485 0.808	1.302 1.290 0.930	1.429 1.435 -0.418	1.096 1.094 0.183	0.435 0.427 1.874
1.301 1.272 2.280	1.486 1.475 0.746	1.315 1.288 2.096	1.505 1.492 0.871	1.294 1.270 1.890	1.342 1.327 1.130	1.395 1.417 -1.553	0.841 0.796 5.653	
1.507 1.481 1.756	1.308 1.279 2.267	1.504 1.489 1.007	1.310 1.287 1.787	1.475 1.479 -0.270	1.305 1.306 -0.077	1.326 1.360 -2.500	0.666 0.667 -0.150	
1.314 1.284 2.336	1.498 1.483 1.011	1.294 1.263 2.454	1.477 1.473 0.272	1.320 1.322 -0.151	1.407 1.419 -0.846	1.008 1.025 -1.659	0.463 0.465 -0.430	
1.494 1.477 1.151	1.307 1.287 1.554	1.344 1.312 2.439	1.308 1.294 1.082	1.412 1.429 -1.190	1.039 1.034 0.484	0.595 0.592 0.507		
1.338 1.328 0.753	1.437 1.442 -0.347	1.399 1.402 -0.214	1.328 1.322 0.454	1.010 1.010 0.000	0.556 0.567 -1.940			
1.262 1.253 0.718	1.104 1.105 -0.090	0.843 0.782 7.801	0.695 0.656 5.945	0.464 0.460 0.870				
	0.536 0.527 1.708	0.407 0.406 0.246						

FIGURE 2.3-17

Waterford Unit 3 Cycle 11 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 465.0 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.873 0.874 -0.114	1.131 1.099 2.912	1.219 1.176 3.656	1.420 1.381 2.824	1.238 1.196 3.512	1.440 1.407 2.345	1.310 1.280 2.344	1.271 1.273 -0.157	0.602 0.614 -1.954
1.131 1.097 3.099	1.272 1.239 2.663	1.402 1.367 2.560	1.227 1.182 3.807	1.424 1.383 2.965	1.272 1.236 2.913	1.402 1.382 1.447	1.131 1.126 0.444	0.519 0.543 -4.420
1.219 1.173 3.922	1.402 1.364 2.786	1.228 1.185 3.629	1.423 1.389 2.448	1.221 1.184 3.125	1.317 1.287 2.331	1.378 1.388 -0.720	0.899 0.874 2.860	
1.420 1.371 3.574	1.226 1.180 3.898	1.423 1.382 2.967	1.245 1.205 3.320	1.425 1.392 2.371	1.270 1.262 0.634	1.330 1.376 -3.343	0.733 0.760 -3.553	
1.238 1.193 3.772	1.424 1.390 2.446	1.220 1.185 2.954	1.426 1.400 1.857	1.284 1.268 1.262	1.373 1.384 -0.795	1.045 1.078 -3.061	0.542 0.567 -4.409	
1.440 1.395 3.226	1.275 1.238 2.989	1.317 1.286 2.411	1.272 1.255 1.355	1.375 1.394 -1.363	1.062 1.066 -0.375	0.671 0.705 -4.823		
1.310 1.283 2.104	1.406 1.396 0.716	1.379 1.378 0.073	1.331 1.330 0.075	1.047 1.060 -1.226	0.622 0.658 -5.471			
1.271 1.274 -0.235	1.137 1.134 0.265	0.901 0.861 4.646	0.770 0.747 3.079	0.542 0.575 -5.739				
	0.604 0.609 -0.821	0.483 0.496 -2.621						

FIGURE 2.3-18

Waterford Unit 3 Cycle 12 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 41.3 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.868 0.883 -1.699	1.093 1.085 0.737	1.165 1.138 2.373	1.478 1.469 0.613	1.215 1.194 1.759	1.480 1.471 0.612	1.182 1.151 2.693	1.453 1.432 1.466	0.681 0.650 4.769
1.093 1.095 -0.183	1.267 1.254 1.037	1.458 1.443 1.040	1.220 1.194 2.178	1.511 1.492 1.273	1.215 1.191 2.015	1.465 1.445 1.384	1.434 1.419 1.057	0.526 0.494 6.478
1.165 1.146 1.658	1.457 1.454 0.206	1.199 1.178 1.783	1.497 1.495 0.134	1.290 1.263 2.138	1.491 1.489 0.134	1.480 1.464 1.093	0.927 0.905 2.431	
1.478 1.457 1.441	1.220 1.194 2.178	1.497 1.483 0.944	1.211 1.196 1.254	1.489 1.497 -0.534	1.215 1.211 0.330	1.431 1.455 -1.649	0.698 0.690 1.159	
1.215 1.195 1.674	1.511 1.507 0.265	1.289 1.268 1.656	1.489 1.494 -0.335	1.183 1.187 -0.337	1.446 1.448 -0.138	0.985 1.004 -1.892	0.520 0.517 0.580	
1.480 1.476 0.271	1.216 1.202 1.165	1.492 1.500 -0.533	1.217 1.210 0.579	1.444 1.446 -0.138	0.961 0.959 0.209	0.571 0.549 4.007		
1.182 1.166 1.372	1.468 1.467 0.068	1.483 1.475 0.542	1.435 1.442 -0.485	0.985 1.009 -2.379	0.570 0.573 -0.524			
1.453 1.464 -0.751	1.436 1.446 -0.692	0.931 0.925 0.649	0.760 0.730 4.110	0.479 0.463 3.456				
	0.682 0.664 2.711	0.529 0.511 3.523						

FIGURE 2.3-19

Waterford Unit 3 Cycle 12 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 258.1 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.838 0.843 -0.593	1.060 1.033 2.614	1.128 1.097 2.826	1.461 1.455 0.412	1.175 1.159 1.381	1.468 1.465 0.205	1.152 1.127 2.218	1.386 1.386 0.000	0.670 0.653 2.603
1.060 1.038 2.119	1.208 1.195 1.088	1.446 1.433 0.907	1.180 1.160 1.724	1.490 1.483 0.472	1.188 1.174 1.193	1.488 1.482 0.405	1.382 1.380 0.145	0.536 0.514 4.280
1.128 1.101 2.452	1.445 1.434 0.767	1.171 1.147 2.092	1.491 1.481 0.675	1.250 1.230 1.626	1.501 1.499 0.133	1.484 1.492 -0.536	0.928 0.923 0.542	
1.461 1.447 0.968	1.180 1.159 1.812	1.491 1.476 1.016	1.180 1.163 1.462	1.482 1.477 0.339	1.172 1.178 -0.509	1.404 1.438 -2.364	0.716 0.721 -0.693	
1.175 1.154 1.820	1.490 1.486 0.269	1.249 1.232 1.380	1.481 1.476 0.339	1.138 1.127 0.976	1.378 1.381 -0.217	0.941 0.960 -1.979	0.541 0.547 -1.097	
1.468 1.465 0.205	1.188 1.179 0.763	1.501 1.508 -0.464	1.173 1.177 -0.340	1.376 1.377 -0.073	0.920 0.905 1.657	0.582 0.559 4.114		
1.152 1.134 1.587	1.489 1.493 -0.268	1.485 1.496 -0.735	1.405 1.425 -1.404	0.940 0.959 -1.981	0.580 0.582 -0.344			
1.386 1.400 -1.000	1.382 1.394 -0.861	0.931 0.938 -0.746	0.771 0.751 2.663	0.501 0.488 2.664				
	0.672 0.660 1.818	0.538 0.529 1.701						

FIGURE 2.3-20

Waterford Unit 3 Cycle 12 Assembly Fr Distribution Comparison Between Plant Measurement and ANC – 495.3 EFPD

Legend: ANC Predicted Fr Power Distribution
Measured Data
% Difference

0.890 0.889 0.112	1.074 1.041 3.170	1.099 1.073 2.423	1.397 1.380 1.232	1.119 1.109 0.902	1.404 1.393 0.790	1.112 1.100 1.091	1.348 1.341 0.522	0.713 0.701 1.712
1.074 1.045 2.775	1.183 1.170 1.111	1.397 1.377 1.452	1.133 1.113 1.797	1.417 1.403 0.998	1.145 1.141 0.351	1.439 1.430 0.629	1.333 1.344 -0.818	0.586 0.572 2.448
1.099 1.075 2.233	1.396 1.374 1.601	1.128 1.108 1.805	1.419 1.401 1.285	1.197 1.178 1.613	1.446 1.436 0.696	1.424 1.441 -1.180	0.944 0.951 -0.736	
1.397 1.374 1.674	1.134 1.111 2.070	1.419 1.398 1.502	1.134 1.120 1.250	1.420 1.408 0.852	1.136 1.156 -1.730	1.369 1.399 -2.144	0.764 0.782 -2.302	
1.119 1.103 1.451	1.417 1.402 1.070	1.196 1.178 1.528	1.420 1.404 1.140	1.112 1.106 0.542	1.349 1.348 0.074	0.966 0.989 -2.326	0.624 0.631 -1.109	
1.404 1.394 0.717	1.144 1.142 0.175	1.446 1.444 0.139	1.137 1.154 -1.473	1.346 1.349 -0.222	0.958 0.942 1.699	0.658 0.637 3.297		
1.112 1.099 1.183	1.439 1.428 0.770	1.426 1.440 -0.972	1.370 1.384 -1.012	0.965 0.988 -2.328	0.657 0.663 -0.905			
1.348 1.342 0.447	1.333 1.347 -1.039	0.947 0.969 -2.270	0.823 0.806 2.109	0.570 0.563 1.243				
	0.714 0.703 1.565	0.588 0.584 0.685						