Proprietary Information - Withhold From Public Disclosure Under 10 CFR 2.390
The balance of this letter may be considered non-proprietary upon removal of Attachments 3 and 6.

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Waterford 3

10 CFR 50.90

W3F1-2018-0031

June 13, 2018

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

SUBJECT: Supplemental Information Supporting the License Amendment Request Regarding Use of the TRANFLOW Code for Determining the Pressure Drops Across the Steam Generator Secondary Side Internal Components
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. W3F1-2018-0014, License Amendment Request for Use of the TRANFLOW code for determining the Pressure Drops Across the Steam Generator Secondary Side Internal Components, April 12, 2018 [NRC ADAMS Accession Number ML18106A074].

2. Waterford Steam Electric Station, Unit 3 – Supplemental Information Needed for Acceptance of Requested Licensing Action Re: Use of the TRANFLOW Code for Determining Pressure Drops Across Steam Generator Secondary Side Internal Components (EPID L-2018-LLA-0112), June 1, 2018 [NRC ADAMS Accession Number ML18145A265].

Dear Sir or Madam:

By letter dated April 12, 2018 (Reference 1), Entergy Operations, Inc. (Entergy) submitted a License Amendment Request pursuant to 10 CFR 50.90 for approval for use of the TRANFLOW code for determining the pressure drops across the steam generator secondary side internal components for Waterford Steam Electric Station, Unit 3 (Waterford 3).

By letter dated June 1, 2018 (Reference 2), the NRC notified Entergy that the NRC staff reviewed this submittal and determined that additional information is necessary to enable the staff to make an independent assessment regarding the acceptability of the proposed amendment in terms of regulatory requirements and the protection of public health and safety and the environment. Reference 2 includes an Enclosure which provides supplemental information requested to make the application complete.

The supplemental information requested by the NRC in Reference 2 is provided in the Enclosure to this letter.

Attached to the Enclosure are:

- LTR-SGMP-18-20 NP-Attachment, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items" (Non-Proprietary) (Attachment 1).

- LTR-SGMP-18-20 P-Attachment, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items" (Proprietary) (Attachment 3).

- LTR-SGMP-17-107 NP-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators" (Non- Proprietary) (Attachment 4).

- LTR-SGMP-17-107 P-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators" (Proprietary) (Attachment 6).

Also attached to the Enclosure are the following:

- Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-18-4763, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice (Attachment 2).

- Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-18-4712, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice (Attachment 5).

As Attachments 3 and 6 contain information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), each is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-18-4763 and/or CAW-18-4712, as applicable, and should be addressed to James A. Gresham, Consulting Engineer, Licensing and Regulatory Affairs, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

This supplement does not alter the no significant hazards consideration or environmental assessment previously submitted by Entergy in letter W3F1-2018-0014 (Reference 1). This supplement does not alter the proposed change to the Waterford 3 licensing basis provided in Reference 1.

There are no new regulatory commitments contained in this supplement.

If you have any questions or require additional information, please contact John Jarrell, Regulatory Assurance Manager, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 13, 2018.

Sincerely,

JCD/JPJ/mmz

Enclosure: Supplement to Proposed Licensing Basis Change. This includes the following 6 attachments:

1. Westinghouse Letter LTR-SGMP-18-20 NP-Attachment, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items," Rev. 1, June 13, 2018. (21 pages)

2. Letter CAW-18-4763, Affidavit for LTR-SGMP-18-20 P-Attachment, Application for Withholding Proprietary Information from Public Disclosure, June 13, 2018. (7 pages)

3. PROPRIETARY – Westinghouse Letter LTR-SGMP-18-20 P-Attachment, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items," Rev. 1, June 13, 2018. (21 pages)

4. Westinghouse Letter LTR-SGMP-17-107 NP-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators," Rev. 0, February 21, 2018. (118 pages)

5. Letter CAW-18-4712, Affidavit for LTR-SGMP-17-107 P-Attachment, Application for Withholding Proprietary Information from Public Disclosure, February 26, 2018. (7 pages)

6. PROPRIETARY – Westinghouse Letter LTR-SGMP-17-107 P-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators," Rev. 0, February 21, 2018. (118 pages)

W/ Attachments 3 & 6

cc: Mr. Kriss Kennedy, Regional Administrator
U.S. NRC, Region IV
RidsRgn4MailCenter@nrc.gov

U.S. NRC Project Manager for Waterford 3
April.Pulvirenti@nrc.gov

U.S. NRC Senior Resident Inspector for Waterford 3
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W/O Attachments 3 & 6

cc: Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
Ji.Wiley@LA.gov

Enclosure to

W3F1-2018-0031

Waterford Steam Electric Station, Unit 3

Supplement to Proposed Licensing Basis Change

(3 pages)

Supplemental Information Requested by the NRC. *Waterford responses are in italics.*

Provide supplemental information/justification supporting the use of TRANFLOW for performing steam generator secondary side internal component pressure drop calculations. The information should include, at a minimum, the following:

1. For the TRANFLOW code:

   - Additional details on the code-to-code benchmarks discussed in Section 3.1 of the submittal, including for each benchmark:
     - a discussion of the event modeled,
     - a comparison of input parameters and nodalizationsand
     - a discussion of how the benchmark analysis is relevant for determining blowdown loads.

     A summary response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 2 of 21.

   - A discussion concerning whether the experimental validation performed in the TRANFLO topical report is relevant to the blowdown loads calculation. If the validation performed in the topical report is relevant, the justification should describe how. If the validation performed in the topical report is not relevant, a comparison of the code to appropriate experiments should be provided and the results discussed and justified to be conservative.

     A summary response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 3 of 21.

     In brief the experimental validation performed in the TRANFLO topical report is directly relevant to the blowdown loads calculation. Qualification of code capability was obtained by numerous predictions of experimental blowdown. TRANFLO has been used to predict the secondary side pressures and mass flow rates during vessel blowdown transients. Details of the blowdown test results are discussed in Section 3.3, Qualification of TRANFLOW vs. Tests of Enclosure 4, (6) LTR-SGMP-107.

     .

   - A discussion detailing any differences between the NRC-approved version of TRANFLO and the TRANFLOW code used in the calculation.

     A summary response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 4 of 21.

     TRANFLO was the original name of the Code and this was renamed to TRANFLOW with migration from a mainframe to a Workstation (UNIX/Linux). The names TRANFLO and TRANFLOW can be used interchangeably when discussing functionality. The original 1974 version of TRANFLO was approved by the NRC in 1976.

2. For the Waterford plant-specific steam generator blowdown load analysis:

   - A detailed discussion of the nodalization used in the steam generator secondary side, steam generator u-tubes, and any components modeled on the primary side of the reactor coolant system that were found to be important to the transient response.

     A summary response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 6 of 21.

   - A discussion of the form loss coefficients applied in the calculation and the basis by which they were developed.

     The response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 11 of 21.

     In brief the Waterford Unit 3 RSG TRANFLOW model the form loss-factors are applied for the flow connectors and segments that include: the flow distribution plate, tube support plates, lower deck plate, primary separator swirl vane blades, secondary separators, steam outlet nozzle, drain pipes, downcomer entrance, wrapper opening at the tubesheet level, etc. These connectors represent flow area changes as a result of SG subcomponents or geometry change.

     The form loss-factors are calculated based on:
     - *Classical methods - Idel'chik Handbook (Reference 8)*
     - *Test data for the primary separators*
     - *Secondary separator TH characteristics provided by the supplier (Peerless)*

   - A discussion of how inputs were biased to ensure that the pressure loading on the internal components was conservative, and how these biases were different from the biases used in the approved application of TRANFLO to determine the flow quality at the break for a main steam line break.

     Conservatism was achieved by adding a 10% bias to steam quality in the original version of TRANFLO. For the WF3 RSG TRANFLOW analysis starting the steam line break transient during hot standby at an initial water level at the top tube support plate results in an increase in differential pressure across the tube support plates of up to 700% compared to starting the transient at the nominal setpoint water level.

     The complete response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 12 of 21.

   - A discussion of how the results of the TRANFLOW blowdown loads analysis are used in downstream mechanical/structural analyses, including a justification for why a simple component pressure drop, such as that provided in Table 1 of the licensee's submittal, is adequate rather than a more detailed distribution of the pressure around each component.

The response can be found on Enclosure 1, (3) LTR-SGMP-18-20 NP (P) starting on page 16 of 21.

Entergy Response
The supplemental information requested by the NRC is addressed by Westinghouse letter, LTR-SGMP-18-20 P/NP-Attachment, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items," Rev. 1, dated June 13, 2018. The letter references LTR-SGMP-17-107 P/NP-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators," Rev. 0, dated February 21, 2018. Applicable documentation is provided as Attachments 1-6 to this Enclosure.

Enclosure Attachment 1 to

W3F1-2018-0031

Westinghouse Letter LTR-SGMP-18-20 NP-Attachment,
"Responses to Waterford Unit 3 TRANFLOW License Amendment Request
Non-Accept Sufficiency Items," Rev. 1

Attachment contains 21 pages

Westinghouse Electric Company

Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items

June 13, 2018

Author: ***Electronically Approved****
Gary W. Whiteman
Plant Licensing and Engineering

Author: ***Electronically Approved****
Jivan G. Thakkar
Steam Generator Management Programs

Verifier: ***Electronically Approved****
David A. Rubolino
Steam Generator Management Programs

Approved: ***Electronically Approved****
David P. Lytle, Manager
Steam Generator Management Programs

**Electronically approved records are authenticated in the electronic document management system.*

I. Introduction

By letter dated April 12, 2018 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML18106A074), Entergy Operations, Inc. submitted a request for U.S. Nuclear Regulatory Commission (NRC) review and approval of the use of the TRANFLOW code to determine the pressure drops across main steam generator secondary side internal components for the plant for Waterford Steam Electric Station, Unit 3.

The NRC staff has evaluated the licensee's submittal relative to the acceptance review criteria provided in subsection 3.1.2 of LIC-109, Revision 2. The NRC staff concluded that:

a) The request used an NRC-approved topical report outside of the limitations imposed by the NRC staff without adequate justification, and

b) The licensee did not provide sufficient information for the NRC staff to complete its detailed technical review (Reference 1).

As a result, supplemental information (i.e., responses to sufficiency items) was requested and must be provided to the NRC staff by June 13, 2018.

Individual responses to each sufficiency item identified in Reference 1 are provided below except additional details on the code-to-code benchmarking discussed in Section 3.1 of the submittal and a discussion whether the experimental validation performed for the TRANFLO topical report is relevant to the blowdown loads calculation. This supplemental information is provided in Reference 2.

Sufficiency Item No. 1 - Provision of Additional Details on Benchmarking of TRANFLO(W)

The supplemental information requested by the NRC staff for benchmarking the TRANFLOW Code is:

- *Additional details on the code-to-code benchmarks discussed in Section 3.1 of the submittal, including for each benchmark:*
  - *Discussion of the event modeled*
  - *Comparison of input parameters and nodalizations*
  - *Discussion of how the benchmark analysis is relevant for determining blowdown loads*

Response:

Additional details on the code-to-code benchmarks discussed in Section 3.1 of the submittal are included in Reference 2.

Four code-to-code benchmark comparisons to TRANFLOW are discussed in Section 3.1 of the submittal: RELAP5, CEFLASH-4A/4B, NOTRUMP and CATHARE 2. Of these comparisons, only the RELAP5 and CEFLASH-4A/4B are relevant for determining blowdown loads.

Additional details on the comparison of the TRANFLOW code and the NRC approved RELAP5 code steam generator simulations can be found in Section 3.1 of Appendix A of

Reference 2. Tube support plate loadings for a small steam line break starting from hot standby for a Model 51 feedwater ring type steam generator are discussed.

Validation of the TRANFLOW code versus the CEFLASH-4B code can be found in Section 3.2 of Appendix A of Reference 2. A comparison of the peak pressure load results during a postulated feedwater line break is provided. Definitions of the TRANFLOW nodes are presented and corresponding nodal locations within the TRANFLOW and CEFLASH-4B models are identified.

The TRANFLOW code is compared to NOTRUMP code in Section 3.5 of Appendix A of Reference 2. Comparisons of water level (narrow range span) and steam pressure responses are shown during a loss-of-normal-feedwater (LONF) flow transient. This comparison demonstrates that TRANFLOW is fully capable of replicating the physics of water level behavior during a LONF transient.

A qualitative comparison of dynamic stability characteristics of steam generators installed at the Cruas Nuclear Power Facility subjected to various levels of tube support plate blockage between the TRANFLOW code and the CATHARE 2 computer code is provided in Section 3.4 of Appendix A of Reference 2. The work presented in this study shows that the CATHARE 2 and TRANFLOW code yield similar results, verifying that TRANFLOW correctly transmits dynamic phenomena and is qualified to simulate dynamic stability responses.

- *A discussion concerning whether the experimental validation performed in the TRANFLO topical report is relevant to the blowdown loads calculation. If the validation performed in the topical report is relevant, the justification should describe how. If the validation performed in the topical report is not relevant, a comparison of the code to appropriate experiments should be provided and the results discussed and justified to be conservative.*

Response:

An accurate prediction of mass and energy release from a vessel means that TRANFLO properly calculates local thermal-hydraulics in various nodes (i.e., elemental control volumes and flow connector).

The experimental validation performed in the TRANFLO topical report is directly relevant to the blowdown loads calculation. Qualification of code capability was obtained by numerous predictions of experimental blowdown. TRANFLO has been used to predict the secondary side pressures and mass flow rates during vessel blowdown transients. Details of the blowdown test results are discussed in Section 3.3, Qualification of TRANFLOW vs. Tests, of Reference 2.

The TRANFLOW computer code has been extensively verified and validated by comparison with other NRC-approved transient analysis codes: RELAP5, TRACE, NOTRUMP, and

CEFLASH-4B. Also, as documented in Reference 2, TRANFLOW has been used to predict the secondary side pressures and mass flow rates during vessel blowdown transients. Reference 2 also includes comparisons between the TRANFLOW calculated values and experiments B53B done at Battelle Northwest, experiments 7, 12, and 14 done at Frankfurt/Main, and several blowdown tests conducted by CISE as part of the CIRENE-3 program. The results obtained from the tests show reasonably good agreement with TRANFLOW calculations. The code has been reviewed and approved by the NRC.

The NRC's review of the TRANFLO code (TRANFLOW is the workstation version of the TRANFLO), included in Section B of Reference 6 (WCAP-8821-P-A), states that:

1. In general, TRANFLO was observed to follow the test data from a number of small blowdown tests (page 7 of Reference 6).
2. Steam generator blowdown simulations were also performed by the NRC using the RELAP-4 (MOD-5) code and a modified version of the COMPARE code. In both the RELAP and COMPARE analyses, the entrainment predicted for full double-ended break was in general agreement with the predictions of the TRANFLO code (page 8 of Reference 6).
3. TRANFLO uses the Moody slip correlation to predict break flow (page 10 of Reference 6). For a large break the results demonstrated that the use of Moody correlation leads to larger flows early in the transient. It is, therefore, concluded that the use of the Moody flow model is satisfactory (page 11 of Reference 6).
4. TRANFLO-MARVEL method is conservative for calculating mass energy release following a main steam line break (page 19 of Reference 6).

Note that the focus of the NRC staff was evaluation of the mass and energy release into a reactor primary containment in the event of a main steam line break.

Based on the favorable comparison of the TRANFLO/TRANFLOW code calculations with other NRC-approved codes and data from several blowdown tests, it is logical to conclude that the TRANFLOW code solves the conservation of mass, momentum and energy equations for the SG transients correctly and has a built in conservative bias. Further details about the acceptability of the TRANFLOW computer code for SG analysis are documented in LTR-SGMP-17-107 (Reference 2).

- *A discussion detailing any differences between the NRC-approved version of TRANFLO and the TRANFLOW code used in the calculation.*

Response:

A discussion detailing any differences between the NRC-approved version of TRANFLO and the TRANFLOW code follows. TRANFLOW Version 3.0 and 3.2 were used in the analysis of the Waterford Unit 3 RSGs**.**

The Original Version (April 1974)

The original homogeneous model predicts mass flow rate, pressure, pressure drop, fluid temperature, steam quality, and void fraction. The code document includes results of TRANFLO calculations for a 51 Series steam generator subject to water and steam blowdown due to a SLB event. The document also presents code verification using blowdown test data from pressurized vessels. Westinghouse documented this version in detail in September of 1976, including code verification using vessel blowdown data (Reference 6).

The Drift-Flux Version (November 1980)

This version implements a drift-flux model to better simulate relative flow velocity between water and steam (Reference 3). For example, it allows a realistic simulation of counter-current flow of steam and water. It required modification of the mass, momentum, and energy equations of the two-phase flow. A capability is provided for monitoring calculated variables for convenient examination of results.

TRANFLO Version 1.0 (November 1991)

This version accepts transient data of parameters as direct inputs, rather than supplying input subroutines, as used in the drift-flux version (Reference 3). It also improves printouts and plots. This version maintains the drift-flux model, and includes the addition of thermal conductivity of Alloy 690 tubing.

TRANFLO Version 2.0 (January 1993)

This version provides as an option for two inlets of feedwater flow into the steam generator (Reference 3). It involves minor changes to a subroutine for specifying feedwater flow. This version is used for separate inlets of simultaneous flow from the main and auxiliary feedwater nozzle.

TRANFLOW Version 3.0 (March 2008)

TRANFLOW is a computer code used to compute the thermal and hydraulic response of steam generators during various design transients in support of the structural qualification of steam generators. This Version 3.0 is developed and configured for the HP-UNIX operating system and has been verified to execute on HP platforms running the HP-UX 11.0 version of the operating systems (Reference 7).

TRANFLOW Version 3.2 (October 2011)

Version 3.2 is developed and configured for the Linux operating system and has been verified to execute on the GNU/Linux 2.6 platforms (Reference 30). This version is functionally identical to TRANFLOW Version 3.0.

Sufficiency Item No. 2 – Discussion of Waterford 3 Plant-Specific SG Blowdown Load Analysis

The supplemental information requested by the NRC staff for discussion of the Waterford Unit 3 plant specific steam generator blowdown analysis is:

- *A detailed discussion of the nodalization used in the steam generator secondary side, steam generator U-tubes, and any components modeled on the primary side of the reactor coolant system that were found to be important to the transient response*

Response:

Information from Reference 4 (CN-NCE-08-44, Revision 1)

The Waterford TRANFLOW model is composed of a network of nodes and connectors that represent the secondary side fluid, tube metal heat transfer, steam generator shell metal heat transfer, tube plate metal heat transfer, and primary coolant. Figures 1 through 3 show the secondary side fluid node / connectors, primary coolant nodes and tube metal heat connectors, and shell metal nodes and connectors. The computational model consists of the following elements:

• Eighteen fluid nodes (#1 - 18) which represent the primary side water volumes.

Primary side volumes are characterized in TRANFLOW as constant pressure nodes with nodal temperatures determined from the energy equation.

• Nineteen fluid node connectors (#1 - 19) between primary side nodes.

Fluid connector number one represents the inlet connector on the hot-leg side; fluid connector number nineteen represents the outlet connector on the cold-leg side. Primary side flow paths are designated in TRANFLOW as boundary connectors wherein all connector parameters (e.g., mass flow rate) remain constant throughout the transient simulation. For boundary connectors, the momentum equation is not solved; hence, pressure drops are not computed.

• Twenty-one fluid nodes (#19 - 39) which represent the secondary side steam and water volumes.

Twenty-one control volumes are used to represent the secondary side of the steam generator. Control volumes nineteen through thirty characterize the entire tube bundle region contained within the wrapper section between the lower deck plate and top surface of the tubesheet. The volume between tube support plates is generally represented by a single control volume, thus providing sufficient detail for accurate assessment of pressure drops across each TSP during faulted and operational design transients. Control volumes thirty-one and thirty-two correspond to the primary riser pipe. Control volume thirty-three represents the upper shell region outside the primary separators between the lower deck and mid deck plates. The steam generator downcomer is represented by three control volumes (thirty-four through thirty-six) extending from the top of the tubesheet to the lower deck plate. Control volumes thirty-seven, thirty-eight, and thirty-nine represent the upper shell region above the mid deck plate.

- Thirty fluid node connectors (#20 - 49) between secondary side nodes.

  Main feedwater coolant is introduced into the steam generator upper shell via fluid connectors forty-seven and forty-eight. These flow connectors represent boundary flow paths in which feedwater temperature and mass flow rate are specified as functions of time. Since spray tubes are uniformly placed along the top surface of the feedwater distribution ring, main feedwater flow is symmetrically introduced into the upper shell. Consequently, the feedwater flow split going into the hot leg (connector forty-seven) and cold leg (connector forty-eight) of the steam generator is fifty-fifty. Steam exits the steam generator outlet nozzle via fluid connector forty-nine.

- Eighteen metal nodes (#1 – 18) which represent the steam generator tube metal.
- Ten metal nodes (#19 – 28) which represent the steam generator shell metal.

  Metal nodes in the steam generator lower shell, transition cone, and portions of the upper shell (below lower deck plate elevation), are divided into two slabs. For inner slabs adjacent to fluid nodes, a half-inch wall layer thickness is assumed. This representation of the steam generator shell is intended to better characterize heat transfer to the steam generator downcomer region during cold auxiliary feedwater addition.

- Four metal nodes (#29 – 32) which represent the tubesheet metal adjacent to secondary fluid. These metal nodes are further subdivided into smaller nodes (#33 – 52). For clarity the subdivided nodes are not shown.
- Fifty-two heat connectors which represent heat transfer links between primary side fluid volume and tube metal, between secondary side fluid volume and tube metal, between steam generator shell and secondary side fluid volume, and between secondary side fluid volume and the top of the tubesheet.

  As indicated in Figures 1 and 2, the fluid node numbering sequence begins on the primary side of the tube bundle (System 1) and then proceeds on the secondary side just above the tubesheet (System 2). This scheme is applicable to non-break as well as main steam line break / feedwater line break simulations. For pipe breaks, fluid node number forty (the last node in the TRANFLOW model) is added and adjustments are made to the appropriate flow connector (either the steam or the feedwater connector depending on which simulation is under evaluation). This numbering scheme eliminates the need to have separate TRANFLOW models for non-break and secondary pipe break transients.

a,c,e

Figure 1
TRANFLOW Model – Steam Generator Secondary Side Fluid Nodes/Connectors

a,c,e

Figure 2
TRANFLOW Model – Primary Coolant Fluid Nodes, Tube/Tubesheet Metal Nodes and Heat Connectors

a,c,e

Figure 3
TRANFLOW Model – Steam Generator Shell Metal Nodes, Fluid Nodes, and Heat Connectors (Non-Steam Line Break)

- *A discussion of the form loss coefficients applied in the calculation and the basis by which they were developed.*

Response:

The form loss coefficient, in a TRANFLOW model, is one of the input parameters for a flow connector that include change in flow areas due to geometry or a presence of a SG internal component. For each segment of the flow connector, values for the positive loss-factor (FDP) and negative loss-factor (FDN) are input entries.

In the Waterford Unit 3 RSG TRANFLOW model the form loss-factors are applied for the flow connectors and segments that include: the flow distribution plate, tube support plates, lower deck plate, primary separator swirl vane blades, secondary separators, steam outlet nozzle, drain pipes, downcomer entrance, wrapper opening at the tubesheet level, etc. These connectors represent flow area changes as a result of SG subcomponents or geometry change.

The form loss-factors are calculated based on:

- Classical methods - Idel'chik Handbook (Reference 8)
- Test data for the primary separators
- Secondary separator TH characteristics provided by the supplier (Peerless)

The following calculation illustrates the tube support plate loss-factor for a TRANFLOW flow connector:

a,c,e

The loss-factor for the flow area contraction and expansion through a tube support plate in the positive flow direction is based on loss-factor through a perforated plate in the I'del'chik handbook.

a,c,e

- *A discussion of how inputs were biased to ensure that the pressure loading on the internal components was conservative, and how these biases were different from the biases used in the approved application of TRANFLO to determine the flow quality at the break for a main steam line break.*

Response:

The SLB thermal/hydraulic analyses of the Delta 110 RSG were performed using the computer program TRANFLOW (Reference 4). Three different cases for SLB conditions were analyzed and are presented in Table 1. All three were initiated from hot standby conditions; one with the water at the top tube support plate, one with the water at the lower deck plate and one with the normal operating water level. The SLB transient is assumed to initiate from hot-standby conditions since this gives the highest possible steam generator operating pressure. With initial conditions at the highest operating steam pressure, blowdown of steam and water through the steam outlet nozzle during a line break is maximized resulting in maximum hydraulic loads on the steam generator internals. The pressure drops across the tube support plates during a postulated SLB event are calculated for each of the three cases and are summarized in Table 1. Figure 4 depicts the tube support plate and wrapper support assembly. Review of these results indicates that the case with initial water level at the top tube support plate produces the greatest SLB loadings on the tube support plates. Tube support plate B sees essentially the same pressure drop with an initial water level at the lower deck plate.

In Table 2 the pressures shown in bold type face for the top tube support plate water level condition in Table 1 are used to determine the enveloping (conservative) forces on the tube support plates (TSPs) shown in Table 2.

By way of contrast, quality is a measure of dryness of the flow, ranging from 1.0 (dry steam) to 0 (saturated liquid). An increment of 0.1 was conservatively added to the qualities predicted by the approved application of TRANFLO (Reference 6).

Table 1 Steam Line Break (SLB) Pressure Drops on Tube Support Plates

Tube Support Plate	Location of Initial Water Level(1),(2)					
	Nominal Setpoint		Lower Deck Plate		Top Tube Support Plate (3)	
	Max (psi)	Min (psi)	Max (psi)	Min (psi)	Max (psi)	Min (psi)
A						
B						
C						
D						
E						
F						
G						
H						

a,c,e

Notes: (1) The load on the tube support plates is produced from a pressure drop across the plate. A positive DP means that the pressure is acting vertically upwards. A negative DP is means that the pressure is acting vertically downwards.

(2) SLB pressure drops per Reference 4.

(3) Values shown in bold bracket all SLB transient cases, the loss of feedwater transient, and the feedwater line transient cases. These values are used as the worst case for the SLB analysis.

Table 2 – Calculation of Vertical Force on the Tube Support Plates Due to SLB Pressure Drops

Tube Support Plate	SLB DP[1] (psi)	Fluid Approach Area[2] (in^2)	Vertical Force Due to SLB DP[3] (lbs)
A			
B			
C			
D			
E			
F			
G			
H			

a,c,e

Notes:
(1) Pressure drops per Table 1 for location of initial water level at the top tube support plate.
(2) Fluid Approach Area per Section 8.1.1of Reference 5
(3) Vertical Force = Fluid Approach Area times SLB ΔP.

a,c,e

Figure 4 – Tube Support Plate and Wrapper Support Assembly –Waterford Unit 3 RSGs

- *A discussion of how the results of the TRANFLOW blowdown loads analysis are used in downstream mechanical/structural analyses, including a justification for why a simple component pressure drop, such as that provided in Table 1 of the licensee's submittal, is adequate rather than a more detailed distribution of the pressure around each component.*

Response:

The TRANFLOW calculated values of thermal-hydraulic (TH) parameters: pressures, pressure loads (ΔPs), flow rates, flow loads (ρV^2), bulk fluid temperatures, metal surface temperatures and film heat transfer coefficients are used in the downstream structural, fatigue and non-ductile failure analyses in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components (Reference 9). **The TRANFLOW calculated TH parameters for each analysis, documented in References 10 through 14, are selected based on the ASME code requirements for the specific evaluation.**

The simple component pressure drop, such as that provided in Table 1, of the licensee's submittal provided a comparison of the pressure differentials across the steam generator (SG) secondary side internal components during a hypothetical steam line break (SLB) transient for the Waterford Unit 3 original steam generators (OSGs) and the replacement steam generators (RSGs).

For the Waterford Unit 3 OSGs, designed prior to the advent of modern computers and advanced computer codes, the pressure differentials (ΔPs) were manually calculated. First the 100% load steady conditions are obtained from the secondary circulation analysis of the OSGs. The secondary side flow rate / velocity during a SLB transient was assumed to be four times greater than at the 100% steady state power. The pressure differential is proportional to the square of the velocity. Therefore, the pressure differentials were calculated to be sixteen times the ΔPs at 100% power conditions.

For the Waterford Unit 3 RSGs these pressure differentials were calculated using the TRANFLOW computer code (Reference 30). Table 1 provides the comparison between the manually calculated values for the OSGs and the TRANFLOW calculated values for the RSGs. The TRANFLOW calculated TH parameters are selected for the downstream analyses as follows:

Use of TRANFLOW TH Parameters in Downstream Mechanical/Structural Analyses

The TRANFLOW calculated TH parameters for each structural analysis are selected based on the ASME code requirements as summarized in the following write-up.

1. As stated in Appendix D of the Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Design Report, Reference 16, the heat transfer coefficient and temperature for the inside of the secondary shell were taken from the TRANFLOW output for the loss of feedwater transient.

2. For performing the thermal analysis of the lower shell and tubesheet in Reference 17, secondary side fluid temperatures and the associated film coefficients were obtained from the TRANFLOW analysis.

3. The temperature time history data used for the boundary conditions in the thermal analyses of the primary manways in Reference 18 were taken from the TRANFLOW outputs.

4. The secondary side pressure time histories, needed for the structural analysis of the tube-to-tubesheet weld in Reference 19, during the cold feedwater at low pressure from hot standby transient were obtained from the TRANFLOW analysis.

5. The temperatures for the Normal/Upset conditions (Level A & B Service Loadings) of the secondary side components for the tubing analysis, in Reference 20, are derived from the TRANFLOW analyses. Also, in this reference, the tube outside wall, secondary side and tube support plate (TSP) temperatures are based on the fluid node temperatures from the TRANFLOW analysis.

6. The inputs, transfer coefficients (HTCs) and bulk fluid temperatures (BFTs), for calculating thermal stress in the lower shell, transition cone and upper shell in Reference 21 are developed from the TRANFLOW Analyses. Also, the governing conditions for the thermal transients are evaluated by TRANFLOW.

7. For performing thermal analysis of small nozzles in Reference 22, the secondary side transient temperatures and pressures are extracted the TRANFLOW data. The film coefficients for the shell side are also extracted from the TRANFLOW data.

8. TRANFLOW calculated thermal-hydraulic parameters for the normal (Service Level A), upset (Level B) emergency (Level C) and faulted (Level D) transients are used for the feedwater nozzle and thermal sleeve evaluations in accordance with the ASME code in Reference 23.

9. The temperatures and pressures for the normal (Service Level A), upset (Level B) emergency (Level C) and faulted (Level D) transients as well as the HTC and BFT inputs for the thermal and structural analyses of the secondary-side manways, in Reference 24, were developed from the Waterford 3 TRANFLOW analyses.

10. The heat transfer film coefficients and bulk steam temperature for the thermal analysis of the steam outlet nozzle and elliptical head assembly, in Reference 25, were extracted from the TRANFLOW analysis.

11. TRANFLOW calculated TH parameters during the feedwater line break, loss of feedwater, and steam line break transients are extensively used for performing the structural analysis, in accordance with the ASME Code, of the internal components in Reference 26. These include:

    a. A unit-loading transient starting from hot standby conditions and ending at full load conditions (i.e., 100% power).
    b. Pressure drops across TSPs.
    c. Vertical loadings for wrapper support system
    d. Wrapper transition cone
    e. Lower deck plate
    f. Mid deck plate
    g. Primary separator swirl vane blades

12. The pressure loads on primary separator components from the TRANFLOW analyses are used in structural analysis of the primary separator assembly. This analysis was performed in accordance with the ASME Code (Reference 27). The loads applied to the primary separator assembly were the limiting cases chosen from the full range of water levels and steam generator tube plugging (SGTP) levels which were analyzed using TRANFLOW code. The flow loads on the primary separator assembly come from pressure drops in the TRANFLOW model.

13. The pressure loads on secondary separator components from the TRANFLOW analyses are used in structural analysis of the secondary separator assembly. This analysis was performed in accordance with the ASME code (Reference 28). The flow loads on the secondary separator assembly come from pressure drops in the TRANFLOW model.

14. TRANFLOW calculated thermal-hydraulic parameters: temperatures, pressure differentials, flow rates and film coefficients for the normal (Service Level A), upset (Level B) emergency (Level C) and faulted (Level D) transients are used for the stress analysis and evaluation of the feedring, feedring supports, and spray nozzles (Reference 29). The structural and fatigue evaluations were performed in accordance with the ASME Boiler and Pressure Vessel Code.

Component Pressure Drop Justification

The total load on each tube support plate due to the net pressure across the plate is calculated based on the fluid approach area to the tube support plate (Reference 5). The area used for the calculation of pressure on the tube support plate is the fluid approach area, not the actual surface area of the TSP. The fluid approach area is used to remain consistent with the pressure data obtained from TRANFLOW, which is calculated based on the fluid approach area. The effect of the trefoil, flow holes and flow slots is accounted for in loss factors within TRANFLOW.

The fluid approach area is calculated as follows:

a,c,e

Multiplying the pressure drop across each tube support plate by the approach area gives a total force on the tube support plate.

References:

1. US NRC Letter from April L. Pulvirenti, Project Manager, to the Site Vice President, Entergy Operations, Inc., “Waterford Steam Electric Station, Unit 3 – Supplemental Information Needed For Acceptance of Requested Licensing Action Re: Use of TRANFLOW Code for Determining Pressure Drops Across Steam Generator Secondary Side Internal Components (EPID L-2018-LLA-0112),” June 1, 2018.
2. Westinghouse Letter LTR-SGMP-17-107 P-Attachment, Revision 0, “Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators,” February 2018.
3. Westinghouse Report WCAP-14046, Revision 3, “Braidwood Unit 1 Technical Support for Cycle 5 Steam Generator Interim Plugging Criteria,” March 1995.
4. Westinghouse Calculation Note CN-NCE-08-44, Revision 1, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Emergency and Faulted Transients to Support Emergency Feedwater System Modifications,” October 2016.
5. Westinghouse Calculation Note CN-NCE-W3RSG-17, “Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Lower Internals Analysis,” October 2012.
6. WCAP-8821-P-A, “TRANFLO Steam Generator Code Description,” June 2001.
7. Westinghouse Letter LTR-NCE-08-3, Rev. 1, “TRANFLOW Version 3.0 User’s Manual,” March 2008.
8. I. E. Idel’chik, “Handbook of Hydraulic Resistances, Coefficients of Local Resistance and of Friction,” Israel Program for Scientific Translation, Jerusalem 1966; Reproduced by National Technical Information Service, U. S. Department of Commerce, Springfield, VA 22161.
9. ASME Boiler and Pressure Vessel Code, Section III, Division 1, “Rules for Construction of Nuclear Power Plant Components,” 1998 Edition with Addenda through 2000, The American Society of Mechanical Engineers, New York, New York.
10. CN-NCE-08-31, Revision 0, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Base Deck Model and Loading/Unloading Transients,” April 2009.
11. CN-NCE-08-42, Revision 0, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Normal Transients,” April 2009.
12. CN-NCE-08-43, Revision 0, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Upset Transients,” June 2009.
13. CN-NCE-08-44, Revision 0, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Emergency and Faulted Transients,” May 2009.
14. CN-NCE-08-44, Revision 1, “Waterford 3 Replacement Steam Generator TRANFLOW Analysis: Emergency and Faulted Transients to Support Emergency Feedwater System Modifications,” October 2016.

15. Not used.
16. WCAP-17066-P, Revision 3, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Design Report," November 2016.
17. CN-NCE W3RSG-1, Revision 1, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Primary Chamber, Tubesheet, Lower Shell and Pedestal Support Complex Analysis," January 2012.
18. CN-NCE-W3RSG-4, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generators Primary Manway Analysis," October 2010.
19. CN-NCE-W3RSG-5, Revision 1, "Waterford 3 Steam Electric Station Model Delta 110 Replacement Steam Generator Tube to Tubesheet Weld Analysis," February 2012.
20. CN-NCE-W3RSG-7, Revision 1, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Tubing Analysis," February 2012.
21. CN-NCE-W3RSG-8, Revision 0, "Waterford 3 Model Delta 110 Replacement Steam Generator Lower Shell, Transition Cone and Upper Shell Analysis," March 2010.
22. CN-NCE-W3RSG-13, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Minor Shell Taps Analysis," November 2010.
23. CN-NCE-W3RSG-14, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Feedwater Nozzle and Thermal Sleeve Analysis," November 2010.
24. CN-NCE-W3RSG-15, Revision 0, "Waterford 3 Steam Electric Station Replacement Steam Generators Secondary Manway Analysis," June 2010.
25. CN-NCE-W3RSG-16, Revision 0, "Waterford 3 Steam Electric Station Model Delta 110 Replacement Steam Generator Steam Outlet Nozzle and Elliptical Head Assembly Analysis," November 2010.
26. CN-NCE-W3RSG-17, Revision 2, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Lower Internals Analysis," October 2012.
27. CN-NCE-W3RSG-18, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Primary Separator Assembly Analysis," November 2010.
28. CN-NCE-W3RSG-19, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Secondary Separator Assembly Analysis," November 2010.
29. CN-NCE-W3RSG-21, Revision 0, "Waterford 3 Steam Electric Station Delta 110 Replacement Steam Generator Feedwater Feedring, Spray Nozzle and Supports Analysis," November 2010
30. LTR-NCE-11-118, Revision 0, "Software Release Letter for TRANFLOW Version 3.2 on GNU/Linux 2.6," Westinghouse Electric Company, October 12, 2011.

Enclosure Attachment 2 to

W3F1-2018-0031

Westinghouse Letter CAW-18-4763,
Affidavit for LTR-SGMP-18-20 P-Attachment
Application for Withholding Proprietary Information
from Public Disclosure

Attachment contains 7 pages

As Attachment 3 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Westinghouse

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USA

U.S. Nuclear Regulatory Commission
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CAW-18-4763

June 13, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4763 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Entergy Operations, Inc.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4763, and should be addressed to James A. Gresham, Consulting Engineer, Licensing and Regulatory Affairs, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

Paul A. Russ

Paul A. Russ, Director
Licensing and Regulatory Affairs

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, Paul A. Russ, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 6/13/18

Paul A. Russ, Director
Licensing and Regulatory Affairs

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

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

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

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

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

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SGMP-18-20 P-Attachment, Revision 1, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items" (Proprietary), for submittal to the Commission, being transmitted by Entergy Operations letter. The proprietary information as submitted by Westinghouse for use by Entergy Operations, Inc. for Waterford Unit 3 demonstrates the acceptability of using the computer code, TRANFLOW, to calculate replacement steam generator secondary side internal loads during a postulated steam line break event.

(a) This information is part of that which will enable Westinghouse to describe the computer code, TRANFLOW, and to benchmark calculated pressure drops using TRANFLOW with other NRC approved computer code results.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse can sell the use of similar information to its customers for the purpose of meeting NRC requirements for licensing documentation supporting the use of the computer code, TRANFLOW, during replacement steam generator design.

(ii) Westinghouse can sell support and defense of this information to customers, if the need arises.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which is utilized by Westinghouse for replacement steam generator design.

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted as attachments to LTR-SGMP-18-20, Revision 1 are proprietary and non-proprietary versions of a document, furnished to the NRC in support of a license amendment to obtain approval to use the computer code, TRANFLOW, for calculating secondary side internal loads during a postulated steam line break event in the Waterford Unit 3 replacement steam generators, and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Entergy Operations, Inc.

Letter for Transmittal to the NRC

The following paragraphs should be included in your letter to the NRC Document Control Desk:

Enclosed are:

1. LTR-SGMP-18-20 P-Attachment Revision 1, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items" (Proprietary)

2. LTR-SGMP-18-20 NP-Attachment Revision 1, "Responses to Waterford Unit 3 TRANFLOW License Amendment Request Non-Accept Sufficiency Items" (Non-Proprietary)

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure CAW-18-4763, accompanying Affidavit, Proprietary Information Notice, and Copyright Notice.

As Item 1 contains information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the items listed above or the supporting Westinghouse Affidavit should reference CAW-18-4763 and should be addressed to James A. Gresham, Consulting Engineer, Licensing and Regulatory Affairs, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

Enclosure Attachment 4 to

W3F1-2018-0031

Westinghouse Letter LTR-SGMP-17-107 NP-Attachment,
"Acceptability of the TRANFLOW Computer Code for
Steam Line Break Internal Pressure Loads for the
Waterford Unit 3 Replacement Steam Generators," Rev. 0

Attachment contains 118 pages

Westinghouse Electric Company

Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators

February 21, 2018

Author: ***Electronically Approved****
Bruce A. Bell for Jivan G. Thakkar
Steam Generator Management Programs

Author: ***Electronically Approved****
Michael J. Sredzienski*
Nuclear Component Engineering-2

Reviewer: ***Electronically Approved****
David A. Rubolino
Steam Generator Management Programs

Approved: ***Electronically Approved****
David P. Lytle, Manager
Steam Generator Management Programs

Electronically approved records are authenticated in the electronic document management system.

I. Introduction

Section 3.9.1.2.2.1.28 of the Waterford Unit 3 Updated Final Safety Analysis Report (UFSAR) incorrectly states that the computer program CEFLASH-4A was used to determine steam generator (SG) internal loads during a postulated steam line break (SLB) event in the design of the Waterford Unit 3 original steam generators (OSGs). It has since been determined by Westinghouse Electric Company (WEC) that CEFLASH-4A was not used. As a result, this issue has been entered into the Waterford Unit 3 corrective action program as CR-WF3-2016-7782 by Entergy Operations.

The primary purpose of this report is to confirm that the computer code, CEFLASH-4A, was not used to calculate postulated SLB internal loads for the Waterford Unit 3 OSGs and to describe the actual method of evaluation used in the original design basis. In addition, this report provides technical justification for acceptability of the computer code, TRANFLOW, for use in the RSG design. This constitutes Task 1a in the WEC offer letter (References 1 and 2).

A review of the analysis of records for Waterford Unit 3 OSGs confirms that pressure loads were manually calculated without the aid of the CEFLASH-4A code. Based on the comparison of results among the CEFLASH, TRANFLOW and manual calculations, the manually calculated loads were conservative.

The use of CEFLASH-4A code in the UFSAR for the Waterford Unit 3 OSGs may have been inadvertently introduced for two potential reasons: (1) CENC-1246 (Reference 3), the primary SG design analysis report for the OSGs lists "CE-FLASH 4" as one of the references and (2) similarity between the Waterford 3 and San Onofre Units 2 and 3 SG designs. All three plants have the same CE-70 series steam generators and share many documents with similar content. Steam generator design and analysis of the Waterford Unit 3 OSGs was followed by the San Onofre OSGs. Based on the available records, the San Onofre Units 2 and 3 OSGs were the first two plants to use CEFLASH-4A for analyzing the SLB transients (References 31 and 32).

A review of the TRANFLOW code qualification reports establishes that the calculations performed with the USNRC approved CEFLASH-4B code and TRANFLOW produce similar results. Westinghouse has used and continues to use the TRANFLOW code for SG design transient analysis. Therefore, the use of the TRANFLOW code for the Waterford Unit 3 RSGs is also acceptable.

II. Waterford Unit 3 Original Steam Generators - Analysis of Records

WEC has reviewed the following documents pertaining to the analysis of records for calculation of the hydraulic loads during the SLB transient:

1. **CENC-1246** (Reference 3): This is the first comprehensive analysis of record (AOR) for the Waterford Unit 3 OSGs. This document establishes the structural integrity of the two Waterford Unit 3 OSGs and describes the SG design in accordance with the requirements in the design specification (Reference 11) and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III for Nuclear Vessels, 1971 Edition and Addenda through Summer 1971 (Reference 4).

In this AOR, the vertical loads resulting from the steam line break (SLB) transient are calculated for the SG components:

A. Baffle (Wrapper) and Baffle Supports (Pages A-374 through A-401)
B. Eggcrates, Tie Rods and Eggcrate Supports (Pages A-447 through A-485)
C. Steam Dryer Supports and Flow Deflector (Pages A-466 through A-479)
D. Tubes and U-Bend Tube Supports (Pages A-541 through A-594)

A. Baffle and Baffle Supports:

The loads on the baffle and baffle supports are calculated based on the loads on:

- Eggcrate Tube Supports
- Can Deck (Lower Deck Plate) and
- Tube Bend Region,

The pressure differentials (ΔPs), for the above components, at 100% load steady conditions are obtained from the secondary circulation analysis of the OSGs (Reference 5). For each component, the 1oad at 100% power is obtained by multiplying the ΔPs by the solid metal area of the component. The total load is then calculated by adding the individual component loads. The total load at four (4) times the 100% power velocity is proportional to the square of the velocity. Therefore, the total pipe rupture load is sixteen (16) times the load at 100% power conditions.

B. Eggcrates, Tie Rods and Eggcrate Supports:

Similar to the baffle and baffle support evaluation, the maximum pressure differential for the eggcrates tie rods and eggcrate supports, during the pipe rupture (SLB) transient, is calculated to be sixteen (16) times the maximum ΔP for the normal operation, 100% power, ΔP.

C. <u>Steam Dryer Support and Flow Deflector:</u>

Based on the summary of results in Section 4.08 of CENC-1246 (Reference 3), the structural and vibration evaluation of the steam dryer support beams and wall brackets considered loadings due to dead weight, steam flow load and flow inducted vibrations. The detailed analysis of these components on Pages A-466 through A-479 in Reference 3 does not mention loads during the SLB transient. As discussed in this AOR, these SG components have significant margins to the ASME Code allowable. The maximum bending stress in the support channel due to normal operating plus design basis earthquake (DBE) loading is 10.0 ksi versus the allowable of 46.7 ksi. The shearing stress on the bolts is 3.0 ksi, which is less than the allowable of 15.6 ksi. In addition, the steam dryers do not constitute a primary fluid (reactor coolant) pressure boundary.

D. <u>Tubes and Tube Supports:</u>

Flow loads are used for performing the stress analysis of the tubes. In the top portion (U-bend region) of the tube bundle, where the tubes are horizontal, the steam-water mixture flows perpendicular to the tubes. The unit flow loads, at 100% power (3410 MWt NSSS), are based on the pressure drop per tube row in the horizontal cross-flow region calculated in Reference 5. During the steam line break (SLB) accident condition, velocity in the tube bundle is assumed to be four (4) times the velocity at 100% power. Therefore, the flow loads on the horizontal sections of the tubes at SLB conditions are sixteen (16) times the load at 100% power.

Similarly, for the stress analysis of the tube supports; the vertical flow load on the tube bundle at accident condition (SLB) is sixteen (16) times the load at 100% power.

Excerpts from CENC-1246 (Reference 3) pertaining to the original hand calculations that were used instead of CEFLASH-4A code are included in Appendix B. Appendix B was added to this report in response to Entergy's comment (Comment #1) based on the review of Revision 0-A of this report. Entergy's reviewer comments, WEC comment resolutions, and the customer reviewer resolution responses are included in Appendix C.

2. **<u>CENC-1512</u>**: "Addendum to the Analytical Report for Louisiana Waterford Unit No. 3 Steam Generator," (Reference 6). This is the second AOR for the Waterford Unit 3 OSGs. This addendum documents changes and additions to CENC-1246 (Reference 3) and addresses the structural integrity of the support skirt, feedwater nozzle, hand-hole cover plate and the gouged area in the as-built upper shell. This report has no relevance to the SLB transient or the CEFLASH-4A computer code.

3. **<u>CENC-1697</u>**: "Addendum to the Analytical Report for Louisiana Waterford Unit No. 3 Steam Generator," (Reference 7). This is the third AOR for the Waterford Unit 3 OSGs. This addendum also documents changes to CENC-1246 (Reference 3) and addresses the structural integrity of the feedwater nozzle and the steam outlet nozzle. This report has no relevance to SLB transient or the CEFLASH-4A computer code.

4. **<u>C-PENG-DR-003</u>**: "Addendum to the Steam Generator Analytical Report for Entergy Operations, Inc. Waterford Unit No. 3," (Reference 8). This is the fourth AOR for the Waterford Unit 3 OSGs. This addendum addressed the effects of a reduction in the feedwater temperature from 70°F to 40°F during the cold feed following hot standby transient in confirming the structural integrity of the

feedwater nozzle. This report has no relevance to the SLB transient or the CEFLASH-4A computer code.

5. **CN-SGDA-03-36**: "Waterford-3 - Steam Generator Structural Evaluation for 3716 MWt Uprate," (Reference 9). The purpose of this report was to confirm that the structural adequacy of the steam generator is maintained for the plant operations at the 3716 MWt Nuclear Steam Supply System (NSSS) power uprate conditions. The steam generator components from CENC-1246 (Reference 3) were reevaluated at the uprated power level.

   The approach for the reevaluation of the steam generator components, in this report, was similar to the original AOR, CENC-1246 (Reference 3). An added steam line break (SLB) load was applied to the SG internals. The applied load was 16 times the change in steady state pressure differential for the component.

6. **SG-SGDA-03-32**: "Addendum to CENC-1246 Analytical Report for Louisiana Waterford Unit No. 3 Steam Generator," (Reference 10). This AOR is also an addendum to the CENC-1246 AOR for addressing the effects of the 3716 MWt NSSS Extended Power Uprate (EPU) Program. This report addresses all components evaluated in Reference 3 for completeness as well as updates results affected by the 3716 MWt EPU program including the reduced steam pressure of 803 psia. This report summarizes conclusions from CN-SGDA-03-36 (Reference 9) and other documents analyzing the impact of the uprated power.

A review of historical design basis for the Waterford Unit 3 original steam generators, from the original design report CENC-1246 compiled in 1975 (Reference 3) and four addenda to the design report (References 7 through 10) prepared through October 2003, was conducted and summarized in this section. These design basis documents reveal that the pipe rupture loads on the OSG internals were calculated assuming that during the SLB transient the secondary fluid velocity in the SG is four (4) times the velocity at normal (100%) power. As a result, the pressure differentials (ΔPs) on the components increase by a square of the velocity or by a factor of sixteen (16). The secondary pressure differentials at 100% power (3410 MWt NSSS power (Reference 11)) steady state conditions, used in the design basis reports, were calculated in 1971 using the CRIB-1 computer code as documented in ST-602, the steam generator circulation report (Reference 5).

In these historical design basis documents for the OSGs, there is no mention of the pipe rupture (steam line break) analysis using any computer code. The hydraulic loads on the SG internal components are manually calculated from the steady state secondary pressure differentials at 100% power level. The review of these historical design basis documents confirms that the Computer Code, CEFLASH-4A (Reference 12), was not used to calculate postulated SLB internal loads for the Waterford Unit 3 OSGs. The loads on SG internals for the CE designed steam generators, subsequent to the Waterford Unit 3 OSGs, were calculated with the transient analysis codes CEFLASH-4A (Reference 12) and CEFLASH-4B (Reference 13). Westinghouse uses the TRANFLOW code (Reference 14) for calculating the loads on the SG internals during the SLB and FWLB transients."

III. TRANFLOW for Waterford Unit 3 RSG Design

Reference 15 provides information about the computer codes used in the Waterford Unit 3 RSG design analysis including TRANFLOW. This section provides technical justification for acceptability of the TRANFLOW computer code (References 14 and 16) for performing RSG design basis analysis including the SLB transient.

WCAP-17066 (Reference 17) is the Waterford Unit 3 Delta 110 Replacement Steam Generator Design Report. This report summarizes the stress analysis results for the RSGs, similar to CENC-1246 (Reference 3) for the OSGs.

CN-NCE-08-44 (References 18 and 19) documents the thermal-hydraulic (TH) analyses for emergency and faulted conditions of the RSGs using the TRANFLOW computer code. These TH analyses include:

- main steam line break (MSLB)
- loss of feedwater (LFW)
- feedwater line break transients (FWLB) and
- TH conditions in the intact (unaffected) SG for the FWLB transient.

Results of the analysis, consisting of hydraulic pressure differentials (ΔPs), were used in the structural qualification of various components inside the steam generators including tube support plates (TSPs), lower and mid-deck plates, primary separators, secondary separators (steam dryers), and the wrapper barrel. As noted previously, the pipe rupture (SLB) loads on the OSG components were manually calculated assuming that ΔPs during the SLB transient across the components increase by a factor of sixteen (16) compared to the ΔPs at 100% power steady state conditions. For the Waterford Unit 3 RSGs, WEC transitioned from manually calculating flow loads during the SLB pipe rupture to the use of the TRANFLOW code.

1. <u>TRANFLOW Computer Code (References 14 and 20)</u>

TRANFLOW is a one-dimensional, two-phase, thermal-hydraulic code used for calculating the thermodynamic and fluid (hydraulic) behavior of steam generators subject to prescribed transient conditions. The code calculates pressures, temperatures, flow rates, and heat transfer coefficients, which are used as inputs for structural analyses. TRANFLOW uses an elemental volume approach in which the spatial solution is achieved by dividing the system into a discrete number of control volumes having uniform thermal and hydraulic conditions. Individual control volumes, or nodes, are interlinked by flow connectors, which are defined by flow area, diameter, length, and hydraulic resistance coefficients. Metal nodes are modeled in TRANFLOW to allow thermal energy storage and heat transfer to the surrounding fluid. Metal nodes are used to represent the steam generator's tubes, shell, and tubesheet. The conservation of mass, momentum, and energy equations are solved in conjunction with boundary conditions specified as functions of time. These boundary conditions include time histories of primary fluid pressure, temperature, and flow rate; steam flow rate, feedwater temperature and flow rate.

TRANFLOW accesses the ASME steam tables for thermodynamic properties and uses a number of empirical heat transfer and fluid flow correlations. Heat transfer between connected fluid nodes and between connected metal and fluid nodes is classified and calculated with the aid of various heat transfer correlations for forced and natural convection; nucleate, transition, and film boiling; and conduction. At each time step, the flow in each connector is compared and limited to a critical or maximum flow for that connector. For subcooled flow, the Zaloudek correlation is used; and for two-phase saturated flow, the Moody correlation defines the critical flow. Effects of velocity differences in two-phase flows are accounted for using a drift flux model formulation.

An early version of TRANFLOW, denoted as TRANFLO, was approved by the USNRC for use with Westinghouse system codes for the calculation of mass and energy releases to containment following a postulated break in a main steam line (References 14 and 29).

2. <u>TRANFLOW Validation and Qualification</u>

The TRANFLOW code has evolved from several transient analysis codes and uses well accepted procedures to calculate SG thermal transient responses. The TRANFLOW code was validated and qualified as follows:

1. Comparisons of TRANFLOW and RELAP steam line break simulations (References 21 and 22). These simulations represented transient behavior of the pressure drop across the Tube Support Plates (TSPs), with the pressure drop being highest at the top TSP during the SLB and decreasing towards the lower TSPs.
2. Comparisons of TRANFLOW and CEFLASH-4B feedwater line break simulations (Reference 23). Steam generator secondary side hydraulic pressure loads across various internals are compared.
3. TRANFLOW blowdown simulations compared to a series of tests performed to qualify the code.
4. TRANFLOW simulation compared to CATHARE 2 code predictions of dynamic stability characteristics of steam generators installed at the Cruas nuclear power facility subjected to various levels of TSP blockage (Reference 24).
5. Comparison of TRANFLOW versus NOTRUMP computer code predictions of the loss-of-normal feedwater transient.
6. Comparison of TRANFLOW versus KSR010 computer code predictions of the turbine trip transient.

More details about the TRANFLOW code and validation are included in Appendix A.

3. <u>Comparison of TRANFLOW Results vs. OSG Methodology for SLB Loads</u>

As discussed previously, the pipe rupture (SLB) loads on the OSG internals were calculated assuming that during the SLB transient the secondary fluid velocity in the SG is four (4) times the velocity at normal (100%) power. As a result, the pressure differentials (ΔPs) across SG components increase by the square of the velocity or by a factor of sixteen (16). Steady state TH performance characteristics of the Waterford Unit 3 RSGs are documented in CN-SGDA-08-16 (Reference 25). The calculation note includes the secondary side pressure drops across the SG internal components at 100% (3739.2 MWt NSSS) power. Table 1 compares steady state ΔPs at 100% power from Reference 25 with the TRANFLOW calculated ΔPs across the SG internals during the SLB transient from Reference 19. The table includes ΔPs and 16 times ΔPs at 100% (3739.2 MWt NSSS) steady state power as well as TRANFLOW calculated ΔPs during the limiting SLB transient.

As presented in Table 1, flow loads with the OSG design methodology (16 times steady state ΔPs) are higher for all internal components except for the secondary separators (steam dryers). The dryers are located nearest the break location (steam outlet nozzle) and therefore experiences initial and more rapid depressurization. Note that as reported in CENC-1246 (Reference 3), the structural and vibration evaluation of the steam dryer support beams and wall brackets had significant margins compared to the ASME Code, Section III allowable (Reference 4). Also note that the steam dryers do not constitute a primary fluid (reactor coolant) pressure boundary.

4. Comparison of TRANFLOW and CEFLASH-4B Results

Both TRANFLOW (Reference 14) and CEFLASH-4B thermal-hydraulic codes (Reference 13 and 26) have been used to calculate pressure differentials across SG internals during postulated pipe breaks, e.g., feedwater line break (FLB), steam line break, and simultaneous feedwater line/steam line breaks. Both TRANFLOW and CEFLASH-4B have been successfully used to evaluate the Westinghouse and ABB/Combustion Engineering fleet of steam generators, respectively. CEFLASH-4A/4B has been accepted by the USNRC for SLB and FLB transients in the pre-certification of the SYSTEM 80^+ NSSS. This was reported to the USNRC in the CESSAR Design Certification (1993). USNRC acceptability of CEFLASH-4B code is also documented in CENPD-252-P-A (Reference 13).

A comparison of the results obtained from the two codes during an FLB accident was conducted and the maximum hydraulic loads for key steam generator components were examined (Reference 23). The comparison of these results is presented in Table 2.

The comparison shows good agreement between the two codes during a rapid secondary side depressurization of a preheat type steam generator. TRANFLOW comparisons to CEFLASH-4B results confirm that TRANFLOW is applicable for different types of steam generator designs, e.g., preheat and feedring type steam generators.

The TRANFLOW model of the preheat SG, definitions of TRANFLOW nodes, comparison of nodal locations within the TRANFLOW and CEFLASH-4B models as well as selective hydraulic load plots comparisons of two codes are included in Reference 23.

5. TRANFLO/TRANFLOW Quality Assurance and Acceptance

The TRANFLO Computer Program was originally developed by MPR Associates, Inc., for Westinghouse in the early 1970s to determine thermal-hydraulic conditions in the steam generators during various transients in order to assist in the design and structural analyses of steam generators (Reference 30).

To demonstrate suitability for this purpose, TRANFLO has been developed and maintained under the Westinghouse Quality Assurance (QA) Program which is in compliance with the design control measures, including verification, stated under 10 CFR 50, Appendix B. Under the Westinghouse QA program, TRANFLO is treated in the same manner as other computer programs used by

Westinghouse in design and safety analyses, including programs that have been submitted to and approved by the USNRC for use in safety analyses. Verification of TRANFLO has included comparison to test data, field data, hand calculations, and independent computer code prediction (Reference 20). TRANFLOW is the workstation version of TRANFLO.

It is WEC's position that the USNRC has implicitly approved the use of TRANFLO/TRANFLOW, as the code has been verified and validated consistent with the Westinghouse Quality Assurance program, and the USNRC has reviewed and approved the elements of that program (Reference 22).

IV. Review of USNRC Approved Codes and Comparison with TRANFLOW

There are several USNRC approved computer codes used in the nuclear industry for analyzing transients in nuclear power plants. A brief review of these codes and comparison of the results with the TRANFLOW code are summarized in this section. More details are provided in Appendix A and the sited references.

1. <u>RELAP5 and Trace</u>

RELAP5 is thermal-hydraulic transient analysis code which has been used as a tool to analyze transients for light water reactor systems. RELAP5 was developed by the Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission.

TRANFLOW code simulations for a postulated SLB transient initiated from hot standby conditions are compared to RELAP5 and the USNRC's TRACE computer codes (References 22 and 21). In References 21 and 22, Westinghouse concluded that SG internal hydraulic loads calculated by the TRACE computer code are comparable to RELAP5 and TRANFLOW. Based on the detailed comparison of SLB results between RELAP5 and TRANFLO/TRANFLOW, for five different SG designs, it was concluded that the two computer codes are technically equivalent (Reference 22).

2. <u>NOTRUMP</u>

NOTRUMP SG models were first used in simulations of the loss-of-normal feedwater and FWLB transients in the late 1970s. The Westinghouse methodology for analyzing FWLB accidents has been accepted by the USNRC on many plant-specific applications. In the early 1980s, NOTRUMP capabilities were expanded to include small break loss-of-coolant accidents (LOCAs) by integrating additional models for break flow, drift flux, core and reactor coolant pumps. Topical report WCAP-10079-P-A (Reference 27) was submitted to the USNRC and in May of 1985, USNRC approved NOTRUMP for calculating small break LOCA events.

The TRANFLOW calculated pressure and SG level responses compares favorably with the NOTRUMP calculations for the Loss-of-Normal Feedwater (LONF) event; defined as a complete loss of main feedwater flow while the reactor is operating at the maximum power level.

3. <u>CEFLASH-4B</u>

As discussed previously in this report, the USNRC acceptability of the CEFLASH-4B code is documented in CENPD-252-P-A (Reference 13). A comparison of the results obtained from the two codes during an FLB accident was performed and the maximum hydraulic loads for key steam generator components were examined (Reference 23). The comparison of these results is presented in Table 2. The comparison shows good agreement between the two codes during a rapid secondary side depressurization of a preheat type steam generator.

Based on these comparisons, it is reasonable to conclude that the TRANFLOW calculated results are in good agreement with other USNRC approved codes. Also, the TRANFLO code (TRANFLOW is the workstation version of the TRANFLO) had been previously accepted by the USNRC Staff for evaluating mass and energy release into a reactor primary containment in the event of a main steam line break

(Reference 29). Westinghouse has used and continues to use the TRANFLO/TRANFLOW computer code for steam generator design transient analysis and the continued use of TRANFLOW in SG design analyses is acceptable.

V. Conclusions

Based on the review of the historical design basis documents for the OSGs and RSGs it is concluded that:

1. A review of the Waterford Unit 3 OSG historical design basis reports (References 3 and 6 through 10) confirm that the CEFLASH-4A computer code was not used for calculating hydraulic loads for the SLB transient.

2. For the Waterford Unit 3 OSGs the loads during the SLB transient were manually calculated based on the pressure differential during normal operation. The loads on the SG internal components, during the SLB transient, are estimated to be sixteen (16) times the steady state pressure differential at 100% power. The factor of sixteen (16) was based on the assumption that the velocity during a pipe rupture (SLB) event is four (4) times higher than at normal steady state operation and the load is proportional to the square of the velocity.

3. For the Waterford Unit 3 RSGs, the loads (ΔPs), during the emergency and faulted conditions (including the SLB transient) documented in References 18 and 19 were calculated using the TRANFLOW computer code.

   As discussed previously, WEC has performed a comparison of TRANFLOW results with those of the NRC approved codes: CEFLASH-4A, CEFLASH-4B and RELAP5. The comparison shows that the results, loads on SG internals during a SLB transient, are technically equivalent. Thus, the NEI 97-06, Revision 3 (Reference 33) steam generator performance criteria were satisfied during a postulated SLB with the use of TRANFLOW in the design of the RSGs. Therefore, the proposed change does not involve a reduction in margin of safety.

4. The comparison between the TRANFLOW and CEFLASH-4B calculated loads (ΔPs) shows good agreement during a rapid secondary side depressurization for the feedwater line break (FLB) transient (Reference 23).

5. As presented in Table 1 of this report, for the Waterford Unit 3 RSGs, the flow loads with the OSG design methodology (16 times steady state ΔPs) are higher than TRANFLOW calculated ΔPs for all SG internal components except for the secondary separators (steam dryers). Note that dryers are located nearest to the break location (steam outlet nozzle) and therefore experience initial and more rapid depressurization. As discussed in Section II of this report, the secondary separators have significant margins to the ASME code allowable. Also note that the steam dryers do not constitute a primary fluid (reactor coolant) pressure boundary.

6. Based on discussions in 4 and 5 above, the manually calculated (16 times steady state ΔPs) SLB loads used in the Waterford Unit 3 OSG historical design basis reports are higher (more conservative) than either CEFLASH-4B or TRANFLOW calculated hydraulic loads (ΔPs).

7. TRANFLO/TRANFLOW calculated results are also in good agreement with other USNRC approved transient analysis codes: TRACE, RELAP5, NOTRUMP and CEFLASH-4B. Therefore, the use of the historical method (16 times steady state ΔPs) is unnecessarily conservative.

8. Westinghouse has used and continues to use the TRANFLOW computer code for steam generator design transient analysis. TRANFLOW has been developed and maintained under the Westinghouse

Quality Assurance (QA) Program which is in compliance with the design control measures, including verification, stated under 10 CFR 50, Appendix B. Therefore, the use of TRANFLOW for the Waterford Unit 3 RSGs is acceptable.

Table 1

Comparison of Steady State and TRANFLOW ΔPs for SLB Transient

Component	Steady State ΔP (Reference 25)	OSG Design Basis Methodology 16 * Steady State ΔP	Max. ΔP - SLB (TRANFLOW - Reference 19, Normal Water Level (WL))	Max. ΔP - SLB (TRANFLOW - Reference 19, All WLs Considered)
	(psi)	(psi)	(psi)	(psi)
TSP - A (Cold Leg Side)	0.082	1.31	-0.65	-1.11
TSP - A (Hot Leg Side)			-0.30	-0.50
TSP - B	0.123	1.97	-0.06	-0.08
TSP - C	0.198	3.17	0.04	0.33
TSP - D	0.282	4.51	0.11	0.70
TSP - E	0.359	5.74	0.21	1.12
TSP - F	0.436	6.98	0.33	1.58
TSP - G	0.515	8.24	0.51	2.09
TSP - H	0.619	9.90	0.75	2.67
U-Bend - Cross Flow	0.689	11.02	-	-
Wrapper Barrel	-	-	7.53	15.17
Wrapper Transition Cone	-	-	8.49	8.49
Lower Deck Plate	1.371	21.94	17.73	17.73
Mid Deck Plate	-	-	3.47	3.47
Secondary Separators (Dryers)	0.136	2.18	7.51	7.59
Primary Separators (Swirl Vane Blades)	1.155	18.48	6.47	6.47

Table 2

Comparison of Peak Pressure Loads for FLB Transient initiated from Full Load (Reference 23)

Location	TRANFLOW Maximum ΔP (psi)	CEFLASH-4B Maximum ΔP (psi)	DEVIATION (%)
Divider Plate near Tubesheet	210	197	6.6
Divider Plate above the Flow Distribution Plate	162	176	7.9
Divider Plate middle of the Preheater Region	142	149	4.7
Divider Plate top of the Preheater Region	79	65	21.5
Divider Plate top of Divider Plate	43	18	139
Flow Distribution Plate	147	129	13.9
Riser to Bottom of Preheater	202	190	6.3

VI. References

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Appendix A

TRANFLOW Computer Code Qualification for the Steam Line Break Analysis of the Waterford Unit 3 Replacement Steam Generators

TRANFLOW Computer Code Qualification for the Steam Line Break Analysis of the Waterford Unit 3 Replacement Steam Generators

1. INTRODUCTION

TRANFLOW is used to calculate heat transfer coefficients for steam generator tubes, shell and tubesheet as well as secondary side fluid temperatures during various design transients. Heat transfer coefficients and secondary side fluid temperatures from TRANFLOW are applied in the heat transfer phase of the structural analysis as boundary conditions in the determination of through-wall temperature gradients and subsequently the resulting thermal stresses. In addition, TRANFLOW is used to predict pressure differentials across tube support plates (TSPs) and other steam generator internal components under steam line break (SLB) conditions in support of the internals design effort.

2. OVERVIEW OF TRANFLOW

TRANFLOW is a one-dimensional, two-phase, thermal-hydraulic code used to calculate the thermodynamic and fluid behavior of steam generators subject to prescribed transient conditions. The code calculates pressures, temperatures, flow rates, and heat transfer coefficients, which are used as inputs for structural analyses. TRANFLOW uses an elemental volume approach in which the spatial solution is achieved by dividing the system into a discrete number of control volumes having uniform thermal and hydraulic conditions. Individual control volumes, or nodes, are interlinked by flow connectors, which are defined by flow area, diameter, length, and hydraulic resistance coefficients. Metal nodes are modeled in TRANFLOW to allow thermal energy storage and heat transfer to the surrounding fluid. Metal nodes are used to represent the steam generator's tubes, shell, and tubesheet. The conservation of mass, momentum, and energy equations are solved in conjunction with boundary conditions specified as functions of time. These boundary conditions include time histories of primary fluid pressure, temperature, and flow rate; secondary side steam flow rate; and feedwater temperature and flow rate.

TRANFLOW accesses the ASME steam tables for thermodynamic properties and uses a number of empirical heat transfer and fluid flow correlations. Heat transfer between connected fluid nodes and between connected metal and fluid nodes is classified and calculated with the aid of various heat transfer correlations for forced and natural convection; nucleate, transition, and film boiling; and conduction. At each time step, the flow in each connector is compared and limited to a critical or maximum flow for that connector. For subcooled flow, the Zaloudek correlation is used; and for two-phase saturated flow, the Moody correlation defines the critical flow. Effects of velocity differences in two-phase flows are accounted for using a drift flux model formulation.

An early version of TRANFLOW, denoted as TRANFLO, was approved by the United States Nuclear Regulatory Commission (USNRC) for use with Westinghouse system codes for the calculation of mass and energy releases to containment following a postulated break in a main steam line (Reference A-1). Specifically, TRANFLOW was used to provide the quality of break flow emanating from the steam line break as a function of time. Quality at the break is important to the overall energy release to containment since liquid ejected from the break has a lower enthalpy than if it were allowed to remain in the steam generator to be boiled by heat from the primary system. Consequently, a lower enthalpy can result in reductions of as much as 10 psi in pressure and 150°F in temperature within the containment when compared to a dry (quality equal to one) steam release (Reference A-1).

TRANFLOW forms the basis for NOTRUMP (References A-2 and A-3), a computer code used in the analysis of various Non-LOCA transient events including (1) Loss-of-Normal Feedwater (LONF), (2) Steam Line Break (SLB), (3) Feedwater Line Break (FLB), and (4) Anticipated Transients without Scram (ATWS). Like TRANFLOW, NOTRUMP is a one-dimensional, two-phase, thermal-hydraulic code used to calculate the thermodynamic and fluid behavior in a nodal network. NOTRUMP contains the same correlations and features as TRANFLOW for simulating important phenomena occurring during steam generator transients. In Section 3.5, TRANFLOW and NOTRUMP predictions are compared for the LONF transient.

3. TRANFLOW VALIDATION AND QUALIFICATION

The TRANFLOW code has evolved from several transient analysis codes and uses well accepted procedures to calculate steam generator (SG) thermal transient responses. The TRANFLOW code was validated and qualified as follows:

1. Comparisons of TRANFLOW and RELAP steam line break simulations (References A-4 and A-5). These simulations represented transient behavior of the pressure drop across the Tube Support Plates (TSPs), with the pressure drop being highest at the top TSP during the SLB and decreasing towards the lower TSPs.
2. Comparisons of TRANFLOW and CEFLASH-4B feedwater line break simulations (Reference A-6). Steam generator secondary side hydraulic pressure loads across various internals are compared.
3. TRANFLOW blowdown simulations compared to a series of tests performed to qualify the code.
4. TRANFLOW simulation compared to CATHARE 2 code predictions of dynamic stability characteristics of steam generators installed at the Cruas nuclear power facility subjected to various levels of TSP blockage (Reference A-7).
5. Comparison of TRANFLOW versus NOTRUMP computer code predictions of the loss-of-normal feedwater transient.
6. Comparison of TRANFLOW versus KSR010 computer code predictions of the turbine trip transient.

3.1 Validation of TRANFLOW vs. RELAP5

TRANFLOW code simulations for a postulated SLB transient initiated from hot standby conditions are compared to RELAP5 and the USNRC TRACE computer codes (References A-4 and A-5). RELAP5 is an advanced thermal-hydraulic transient analysis code which has been used as a tool to analyze transients for light water reactor systems. RELAP5 was developed by the Idaho National Engineering Laboratory for the U.S. Nuclear Regulatory Commission.

In Reference A-5, Westinghouse concluded that SG internal hydraulic loads calculated by the USNRC's TRACE computer code are comparable to RELAP5 and TRANFLOW. The comparison of the SLB TSP pressure drops (hydraulic loads) obtained using the TRANFLOW and RELAP5 are presented in Table 3.1-1. TSP pressure drop values are plotted in Figure 3.1-1. During an SLB, blowdown of steam and water through the steam line break leads to a rapid depressurization of the secondary side. In response to this depressurization, the fluid in the tube bundle undergoes a rapid acceleration leading to the development of internal pressure drops which exert hydraulic loads on the tube support plates. Maximum pressure drops, as a result, are developed for the topmost TSPs. Note: Positive TSP pressure drops are directed vertically upwards and negative pressure drops are directed vertically downwards.

In general, the pressure drop comparisons of the TSPs are reasonably good. The maximum deviation between individual support plate loadings is only 0.27 psi (1.9 kPa) which is for TSP Number 3.

Table 3.1-1

Tube Support Plate Loading Comparison between RELAP5 and TRANFLOW Steam Line Break

Tube Support Plate Number, from Bottom to Top	**RELAP5 Loads from Figure 4-7 in Reference A-4**		**TRANFLOW Case 1 Loads from Figure 4-12 in Reference A-4**		**Deviation**	**Deviation**
	psi	**bar**	**psi**	**bar**	**bar**	**(%)**

a,c,e

Conclusions from comparing the results between RELAP5 and TRANFLOW in Table 3.1-1 are as follows:

- The peak loading on the top tube support plate is ~9% lower (0.014 bar lower) using RELAP5 compared to using TRANFLOW.
- The maximum absolute difference between individual support plate loadings is only 0.019 bar (for TSP 3).
- Peak loading on the top tube support plate combined with the additional upward loadings on the other support plates (at the time of peak loading of the top TSP during the SLB transient) is 7% higher using RELAP5/MOD3 compared to TRANFLOW.

Based on this comparison, it is concluded that the two computer codes are technically equivalent in their predictions of TSP loadings during an SLB.

Figure 3.1-1

RELAP5 versus TRANFLO Tube Support Plate Loadings for Small Steam Line Break from Hot Standby, Model 51 Feedwater Ring Type Steam Generator

a,c,e

3.2 Validation of TRANFLOW vs. CEFLASH-4B

Both TRANFLOW and CEFLASH-4B thermal-hydraulic codes (Reference A-8) have been used to calculate pressure differentials (ΔPs) across SG internals during postulated pipe breaks, e.g., feedwater line break (FLB), steam line break (SLB), and simultaneous feedwater line/steam line breaks (FLB/SLB). TRANFLOW and CEFLASH-4B have been successfully used to evaluate the Westinghouse and Combustion Engineering fleet of steam generators, respectively. CEFLASH-4B has been used for many years on various types of steam generators for determining SLB and FLB loads on SG internals. CEFLASH has been accepted by the USNRC for SLB and FLB transients in the pre-certification of the SYSTEM 80^+ NSSS. This was reported to the USNRC in the CESSAR Design Certification (1993). USNRC acceptability of CEFLASH-4B code is also documented in CENPD-252-P-A (1979).

A comparison of the results obtained from the two codes during an FLB accident was conducted and the maximum hydraulic loads for key steam generator components were examined (Reference A-6). The comparison of these results is presented in Table 3.2-1. The TRANFLOW model of the preheat SG is shown in Figure 3.2-1. Definitions of TRANFLOW nodes are presented in Table 3.2-2. This table provides the corresponding nodal locations within the TRANFLOW and CEFLASH4B models.

Selective hydraulic load plots comparisons of TRANFLOW versus CEFLASH-4B are provided in Figures 3.2-2 through 3.2-5 for various SG components. The comparison shows reasonably good agreement between the two codes during a rapid secondary side depressurization of a preheat type steam generator. TRANFLOW comparisons to CEFLASH-4B results provide added confirmation that TRANFLOW can handle different types of steam generator designs, e.g., preheat and feedring type steam generators.

Table 3.2-1

Peak Pressure Loads for FLB Transient initiated from Full Load

LOCATION	TRANFLOW MAX ΔP (PSI) (kPa)	CEFLASH MAX ΔP (PSI) (kPa)	DEVIATION (%)
Divider Plate near Tubesheet	210 (1447.9)	197 (1358.3)	6.6
Divider Plate Above the Flow Distribution Plate	162 (1116.9)	176 (1213.5)	7.9
Divider Plate Middle of the Preheater Region	142 (979.1)	149 (1027.3)	4.7
Divider Plate Top of the Preheater Region	79 (544.7)	65(448.2)	21.5
Divider Plate Top of Divider Plate	43 (296.5)	18 (124.1)	139
Flow Distribution Plate	147 (1013.5)	129 (889.4)	13.9
Riser to Bottom of Preheater	202 (1392.7)	190 (1310.0)	6.3

Figure 3.2-1

TRANFLOW Model – Steam Generator Secondary-Side Fluid Nodes/Connectors

a,c,e

Table 3.2-2

TRANFLOW and CEFLASH4B Models Corresponding Nodes

a,c,e

TRANFLOW NODE	CEFLASH NODE(S)	DESCRIPTION OF TRANFLOW FLUID NODE[1]

Note: ATSG denotes "advanced tube support grid"; FDB denotes "flow distribution baffle."

Figure 3.2-2
Steam Generator FWLB – 100% Power
Pressure Difference TRANFLOW Nodes (31, 30)

a,c,e

Figure 3.2-3
Steam Generator FWLB – 100% Power
Pressure Difference TRANFLOW Nodes (33, 32)

a,c,e

Figure 3.2-4
Steam Generator FWLB – 100% Power
Pressure Difference TRANFLOW Nodes (35, 34)

a,c,e

Figure 3.2-5
Steam Generator FWLB – 100% Power
Pressure Difference TRANFLOW Nodes (37, 38)

a,c,e

3.3 Qualification of TRANFLOW vs. Tests

Qualification of code capability is also obtained by means of numerous predictions of experimental blowdown. TRANFLOW has been used to predict the secondary side pressures and mass flow rates during vessel blowdown transients. Included are Experiments B53B done at Battelle Northwest, Experiments 7, 12, and 14 done at Frankfurt/Main, and several series of blowdown tests conducted by CISE as part of the CIRENE-3 program (Reference A-1). Figures 3.3-1 through 3.3-4 depict comparisons of Frankfurt Test 7 to TRANFLOW calculations for various thermal-hydraulic parameters. Favorable comparisons between the mass and pressure transients for the Battelle Northwest and TRANFLOW calculations were obtained as shown in Figures 3.3-5 through 3.3-7. Figures 3.3-8 and 3.3-9 depict comparisons of the CISE test to TRANFLOW calculations for various thermal-hydraulic parameters. In general, the results obtained from the tests show reasonably good agreement with TRANFLOW calculations.

The three Frankfurt/Main tests (7, 12 and14) as mentioned in this section, refer to three water levels within the test vessel. The Frankfurt/Main test vessel is shown in Figure 3.3-10. The three respective water levels relate to 1) Test 7 water below the break level; 2) Test 12 water level above the break point and 3) Test 14 water level is considered near the top (almost full). The figures from Test 7 represent the secondary mass flow rate discharge and the pressure pulse at the break location as a function of time, respectively (Figures 3.3-1 and 3.3-2). Figures 3.3-3 and 3.3-4 present the pressure and specific energy as a function of mass flow rate at the break point.

The Frankfurt/Main Test 7 results compared to TRANFLOW are shown in Figures 3.3-1 through 3.3-4. The TRANFLOW calculated values identified above are generally conservative based on the area under the curve. Experience has shown that the peak hydraulic loads on the SG internals during the steam line break event occur early during the transient and are clearly depicted in the curves. Also, Figure 3.3-4 shows that predicted TRANFLOW mass/energy release is conservative as compared to the test data. Similar results for Frankfurt/Main Test 12 and Test 14 are presented in Reference A-1.

Battelle Test B53B results compared to TRANFLOW are presented in Figures 3.3-5 through 3.3-7 and correspond to liquid level remaining in the vessel, weight of water mass and pressure in the vessel as a function of time during the depressurization transient. The Battelle B53B test vessel is shown in Figure 3.3-11.

The CISE test results compared to TRANFLOW are presented in Figures 3.3-8 and 3.3-9 and correspond to weight of water mass and pressure in the vessel as a function of time during the blowdown transient. The CISE test vessel is shown in Figure 3.3-12.

The TRANFLOW calculated break flow rate and rate of steam pressure decrease are conservatively higher than the measured values during the first few seconds when peak hydraulic loads are typically developed within the steam generator. Greater break flow and rate of fluid depressurization results in rapid fluid motion which results in high pressure drops and hydraulic loads across the tube support plates and other internal structures.

Figure 3.3-1

Comparison of Calculated and Measured Mass Flow Rate for Frankfurt Test 7

a,c,e

Figure 3.3-2

Comparison of Calculated and Measured Pressure for Frankfurt Test 7

a,c,e

Figure 3.3-3

Comparison of Calculated and Measured Pressure vs. Mass Flow for Frankfurt Test 7

a,c,e

Figure 3.3-4

Comparison of Calculated and Measured Energy vs. Mass Flow for Frankfurt Test 7

a,c,e

Figure 3.3-5

Liquid Level of Fluid Remaining in Vessel, Battelle Test B53B

a,c,e

Figure 3.3-6

Weight of Water in Vessel as Function of Time (sec), Battelle Test B53B

a,c,e

[ ]

Figure 3.3-7

Comparison of Calculated and Measured Pressure, Battelle Test B53B

a,c,e

Figure 3.3-8

CISE Blowdown Test 1 Mass vs. Time

a,c,e

Figure 3.3-9

CISE Blowdown Test 1 Pressure vs. Time

a,c,e

Figure 3.3-10

Frankfurt/Main Test Vessel Schematic

a,c,e

Figure 3.3-11

Battelle B53B Test Vessel Schematic

a,c,e

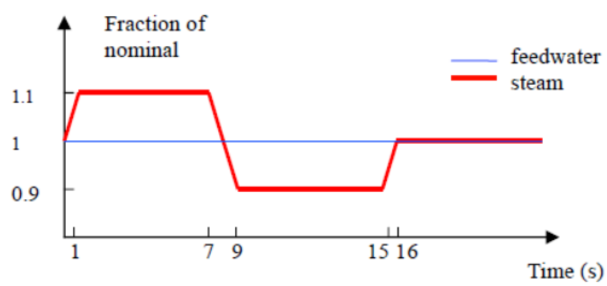
Figure 3.3-12

CISE Test Vessel Schematic

a,c,e

3.4 Qualitative Comparison of Dynamic Stability Characteristics of the Cruas and Model 80F Steam Generators

In a paper presented at the 16th International Conference on Nuclear Engineering (ICONE16) in Orlando, Florida, USA, Vergnault et al., of the French Institute for Radiological Protection and Nuclear Safety (IRSN) discussed the dynamic stability characteristics of steam generators installed at the Cruas nuclear power facility subjected to various levels of tube support plate (TSP) blockage (Reference A-9). In the IRSN study, a stand-alone model of the steam generator was evaluated for dynamic stability using the CATHARE 2 computer code. Boundary conditions consisted of a 10-percent step increase-decrease in the steam flow rate with the feedwater flow rate held constant as shown in the following figure:



The major findings of the IRSN study are shown in Figure 3.4-1; the top two figures show SG pressure and water level responses for the Cruas SGs. For a clean steam generator with no TSP blockage, variations of pressure and water level were non-periodic with a transitory response that quickly returned to nominal values shortly after restoration of steam flow. In the case with no TSP blockage, the Cruas steam generator operates in a very stable manner. With eight-percent (8%) blockage added to the baseline TSP blockage distribution, oscillations in pressure and water level are observed and persist over the 100 second time frame. With ten-percent (10%) blockage added to the baseline TSP blockage distribution, the oscillations were larger in magnitude. The peak-to-peak variation in pressure is approximately two bars and the corresponding SG level deviation is approximately 25% of the Narrow Range (NR). Oscillations corresponding to ten-percent additional blockage are clearly indicative of unstable SG operation.

In order to confirm the general trends of dynamic stability with TSP blockage as observed in the IRSN study, a Model 80F SG was subjected to the identical "stability transient" defined by a 10-percent step increase-decrease in steam flow (feedwater flow held constant). Blockage was limited to the top TSP and three levels were considered, namely, 0%, 72.5% and 77.5% reductions in top TSP flow area. The 72.5% reduction in flow area represented a level of blockage just below the threshold for producing instabilities. The 77.5% reduction in flow area represented a level of blockage that is at or above the threshold for producing steam generator instabilities.

Operating conditions for TRANFLOW Dynamic Stability Analysis are provided in Table 3.4-1.

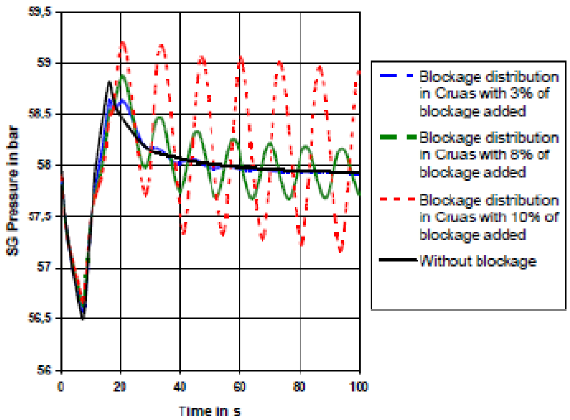
Table 3.4-1
Operating Conditions for TRANFLOW Dynamic Stability Analysis

Parameter	Value
Thermal Power per SG	954.25 MWt
Primary Side Pressure	15.5 MPa
Primary Side Average Temperature	306.5°C
Primary Side Flow Rate	23840 m^3/hr
Steam Dome Pressure	6.9 MPa
Feedwater Temperature	227°C
Circulation Ratio	4.6
Tube Plugging	0%

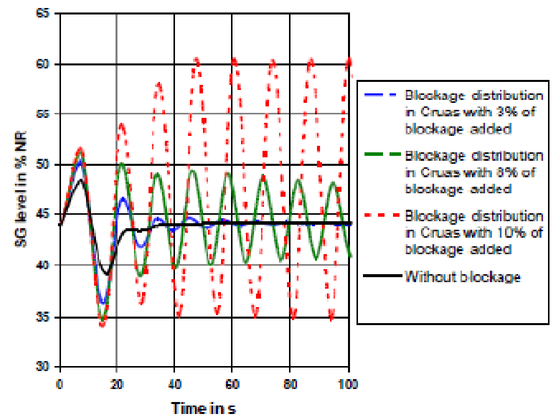
Comparisons of results are shown in Figure 3.4-1; the bottom two figures show SG pressure and water level responses for the Model 80F RSGs. In the case of no blockage (0.0% reduction in top TSP flow area), variations of pressure and water level were non-periodic with pressure and water level transitioning to values existing at the start of the transient by approximately 100 seconds. In the case of no blockage, the Model 80F RSG operates in a very stable manner. With 72.5% reduction in top TSP flow area, a damped oscillatory pressure-water level response is observed which approaches values existing at the start of the transient. Finally, with 77.5% reduction in top TSP flow area, the steam generator is clearly unstable, with large, persistent variations in pressure and water level.

Comparison of the Model 80F to the Cruas steam generators shows qualitatively similar pressure-water level responses for varying TSP blockages ranging from no blockage to high blockage level corresponding to unstable SG operation. When there is no TSP blockage, the Model 80F and Cruas steam generators operate in a very stable manner with variations of pressure and water level quickly returning to nominal values existing at the start of the transient. Moreover, when there is no blockage, the pressure-water level response is non-periodic for both SGs. The work presented in this study shows that CATHARE 2 and TRANFLOW yield similar results, verifying that TRANFLOW correctly transmits dynamic phenomena and is qualified to simulate dynamic stability transients.

Figure 3.4-1: Qualitative Comparison of Dynamic Stability Characteristics of the Cruas and Model 80F Steam Generators



IRSN Simulation of Cruas Steam Generators: SG Pressure Response



IRSN Simulation of Cruas Steam Generators: SG Water Level Response

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TRANFLOW Simulation of Model 80F RSGs: SG Pressure Response

TRANFLOW Simulation of Model 80F RSGs: SG Water Level Response

3.5 Validation of TRANFLOW versus NOTRUMP Computer Code based on Loss-of-Normal Feedwater Transient

In this section, TRANFLOW is compared to NOTRUMP, a Westinghouse proprietary code for the evaluation of steam generator operational transients. The comparison is based on the loss-of-normal feedwater (LONF) transient.

Description of NOTRUMP Computer Code

NOTRUMP is a transient thermal-hydraulic code that models steam generators in pressurized water reactor (PWR) plants (Reference A-2). The code is used to evaluate various non-LOCA (loss-of-coolant accident) events such as loss-of-normal feedwater (LONF), steam line break (SLB), feedwater line break (FLB) and anticipated transients without scram (ATWS). NOTRUMP is used to calculate break quality (SLB, FLB), degradation of SG heat transfer due to reduction in secondary side fluid mass (LONF, FLB, ATWS) and to correlate SG fluid mass with narrow range water level (LONF, FLB). Important phenomena modeled in NOTRUMP include single- and two-phase flow, heat transfer through the tube bundle, critical flow, and special SG separator models for mechanical separation of steam-water.

NOTRUMP, like TRANFLOW, uses an elemental volume approach in which the spatial solution is achieved by dividing the system into a finite number of regions (control volumes). Individual control volumes are linked by flow connectors (links) which are defined by flow area, diameter, length and hydraulic resistance coefficients. Within a control volume, NOTRUMP can simulate either a single homogeneous region (like TRANFLOW), or two separate regions with a distinct mixture level. In the latter representation, the top node consists of vapor while the bottom node consists of a two-phase mixture. The volume of each separate region can vary with time and the regions can be in thermal non-equilibrium with each other. The thermal-hydraulic equations consist of separate mass and internal energy balances for each region, a mixture momentum equation, plus an algebraic drift-flux correlation for the flow connectors. To prevent unrealistic layering of vapor and two-phase mixtures in adjacent vertical-control-volumes, node stacking capability is available for calculating a single mixture elevation. Additional special purpose models include flooding, bubble rise, continuous contact flow link, variable area flow links and a horizontal stratified flow model.

NOTRUMP Code Qualification

NOTRUMP SG models were first used in simulations of the loss-of-normal feedwater and feedwater line break transients in the late 1970s. The use of NOTRUMP was first presented to the USNRC for the analysis of the FLB event in Reference A-10. This topical report was submitted to the USNRC as the licensing basis for analyzing feedwater line break accidents. In these analyses, primary-to-secondary heat transfer, narrow range water level versus SG mass, and break flow quality were calculated by NOTRUMP for use in Westinghouse primary systems code simulations. Although several rounds of Request for Additional Information (RAIs) were exchanged with the USNRC, the USNRC never issued a specific Safety Evaluation Report (SER) on Reference A-10. However, the Westinghouse methodology for analyzing feedwater line break accidents has been accepted by the USNRC on many plant-specific applications.

In the early 1980s, NOTRUMP capabilities were expanded to include small break LOCAs by integrating additional models for break flow, drift flux, core and reactor coolant pumps. Topical report

WCAP-10079-P-A (Reference A-2) was submitted to the USNRC and in May of 1985, the USNRC approved NOTRUMP for calculating small break LOCA events.

Loss-of-Normal Feedwater Flow Transient Simulation in TRANFLOW and NOTRUMP

The Loss-of-Normal Feedwater (LONF) event is defined as a complete loss of main feedwater flow while the reactor is operating at the maximum power level. A loss of main feedwater flow may occur due to the following causes:

- Breaks in the main feedwater system piping upstream of the main feedwater check valves (feedline breaks downstream of the check valves are considered in the Feedline Break analysis).
- Failure or trip of the main feedwater pumps, including loss of power (for motor-driven feedwater pumps) or loss of motive steam (for turbine-driven feedwater pump).
- Spurious closure of the main feedwater isolation valves or the main feedwater regulating valves.

The analysis of the loss-of-normal feedwater flow transient is based on the following assumptions:

1. From steady-state, feedwater flow is assumed lost starting at zero seconds.
2. Feedwater flow goes to zero in 0.1 seconds.
3. Primary side flow, steam flow and feedwater temperature are held constant for the duration of the transient simulation.
4. Primary inlet temperature (T_{hot}) increases from 626.2^{o}F (330.1^{o}C) at zero seconds to 646.1^{o}F (341.2^{o}C) at eighty seconds; afterwards primary inlet temperature is held constant at 646.1^{o}F (341.2^{o}C).
5. No credit is taken for reactor trip (or turbine trip) during the transient.

Key geometric and thermal-hydraulic operating conditions are summarized in Table 3.5-1.

Table 3.5-1

Key Geometric and Operating Conditions for Loss-of-Normal Feedwater Flow Transient

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Parameter	Value

Comparisons of water level (narrow range span) and steam pressure responses are shown in Figure 3.5-1. Following the loss of feedwater flow, steam generator water level drops at a fairly uniform rate until the lower pressure tap is uncovered at approximately 40 seconds. Afterwards, NOTRUMP shows a sudden drop in water level followed by a short recovery. At approximately 60 seconds, NOTRUMP indicates a very erratic response in level. By this time, SG water level has fallen within an area where low-low level trip setpoints exist (less-than 20% narrow range span) thereby producing a reactor trip signal. As shown in Figure 3.5-1, the TRANFLOW level response compares favorably to that predicted by NOTRUMP. While TRANFLOW does not predict a sudden drop in SG water level at approximately 40 seconds, the water level falls below 20% narrow range resulting in nearly identical times for the reactor trip signal. Note: TRANFLOW prematurely terminates at around 64 seconds as a result of low SG mass. Steam pressure responses are also comparable for the loss-of-normal feedwater transient. Steam pressure initially increases as a result of increases in primary coolant temperatures. At approximately 50 seconds, steam pressure reaches a maximum. Afterwards, pressure decreases as a result of degraded primary-to-secondary heat transfer from lack of SG mass. This comparison demonstrates that TRANFLOW is fully capable of replicating the physics of level behavior during the loss-of-normal feedwater transient.

Figure 3.5-1: TRANFLOW and NOTRUMP Comparisons of SG Water Level and Steam Pressure Responses during Loss-of-Normal Feedwater Transient

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3.6 Validation of TRANFLOW versus KSR010 Computer Code based on Turbine Trip Transient

In this section, TRANFLOW is compared to KSR010, a Westinghouse proprietary code for the evaluation of operational transients. The comparison is based on the turbine trip transient.

Description of KSR010 Computer Code

KSR010 is a transient thermal-hydraulic code that models Westinghouse pressurized water reactor (PWR) plants with feedring type steam generators (Reference A-11). The code is used for operational transient analyses with emphasis on steam generator level response during load rejections, loss of feedwater pumps, power increases/decreases, etc. Plant components represented include the steam generator, condensate and feedwater systems, steam line-to-steam header, a simplistic reactor coolant system with nuclear core kinetics (point-kinetics model) and automatic NSSS control systems (reactor control, steam bypass and steam generator level). The condensate and feedwater model includes the feedwater pumps, feedwater control valves and feedwater dynamic-elevation head losses. The heater drain and condensate system performance is simulated by boundary conditions at the suction of the main feedwater pumps. KSR010 is coded in the ACSL (Advanced Continuous Simulation Language) dynamic-programming language, a FORTRAN based language with additional modeling features which allow the changing of input parameters during transient simulations.

The KSR010 steam generator model uses an elemental control volume approach in which the spatial solution is achieved by dividing the SG into a number of regions consisting of (1) primary side tubes, (2) secondary side tube bundle area, (3) riser section extending from bundle exit through the primary separators, (4) upper SG downcomer extending from start of shell transition cone to top of primary separators, (5) straight cylindrical portion of SG downcomer below shell transition cone and (6) steam dome above top of primary separators. Within the tube bundle region, the boiling height is determined by representing the transition from subcooled to nucleate boiling. Separate mass and energy balances are performed for each region in order to determine thermal-hydraulic properties. An overall momentum balance is performed to calculate the change in regional mass flow rates. The momentum balance includes detailed representations of friction and form losses within the downcomer, bundle and riser segments. In addition, dynamic pressure losses are included for developing pressures at the feedwater nozzle inlet and steam generator exit.

SG water level control is simulated in KSR010 for both low and high power modes. The high power mode controls the main feedwater control valve, while the low power mode controls the bypass control valve. The level controller is based on a three element proportional-integral control of SG water level with measurements of narrow range level, steam flow and feedwater flow. The feedwater pump speed controller is based on logic that compares the difference between feedwater and steam header pressures to a DP setpoint for a given steam flow rate. Feedwater pump speed is controlled via a proportional-integral-controller based on the difference between the measured (calculated) and setpoint DP.

KSR010 Code Validation

KSR010 has been validated by comparing code predictions of SG water level to plant data for large 75%-25% load rejection and loss-of-feedwater pump transients (Reference A-12). For the large load reduction transient, KSR010 results compared very well to the measured SG level data, especially for the initial 50 to 60 seconds. The effects of plant nonlinearities or non-idealisms were more pronounced in the long term (greater than 60 seconds), especially for spikes in steam flow that were not properly portrayed by KSR010. However, it was judged that the KSR010 response was acceptable for the analysis of load rejections of concern to steam generator level performance. For the loss-of-feedwater flow transient, the minimum SG level predicted by KSR010 was somewhat lower than the plant measured value, which is conservative. Moreover, the timing of the minimum SG level, peak in T_{avg} and level overshoot following reduction of steam flow below feedwater flow were similar. Therefore, it was concluded that the KSR010 code or its later version, KSR011, is acceptable for the analysis of operational condition transients. KSR011 was created to better handle a reactor trip transient.

Turbine Trip Transient Simulation in TRANFLOW and KSR010

The turbine trip simulation is based on the following assumptions:

- Initial steady-state run of ten seconds
- Turbine trip initiated at ten seconds
- Reactor trip occurs on turbine trip; initiated two seconds after turbine trip
- No credit is taken for steam dump valves or SG power operated relief valves (PORVs)
- Feedwater flow maintained in automatic control after the turbine trip (no loss of feedwater)
- Steam pressure rises to the SG safety valve setpoint, after which the valve opens

Key geometric and thermal-hydraulic operating conditions are summarized in Table 3.6-1:

Table 3.6-1

Key Geometric and Operating Conditions for Turbine Trip Transient

Parameter	Value

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Critical parameter responses predicted by KSR010 are shown in Figure 3.6-1 and include nuclear power, turbine load, T_{avg}, T_{ref} (reference temperature), T_{hot}, T_{cold}, steam flow rate and feedwater flow rate. In the TRANFLOW simulation, T_{hot}, steam flow rate and feedwater flow rate were input as boundary conditions along with feedwater temperature (not shown in Figure 3.6-1). Comparisons of water level (narrow range span) and steam pressure are shown in Figure 3.6-2. Following turbine trip at ten seconds, T_{avg} initially rises and shortly afterwards falls due to the succeeding reactor trip. Steam pressure increases rapidly and levels out when the steam generator safety valves open. As indicated, SG water level drops rapidly after reactor trip even though feedwater flow is maintained higher than steam flow. This response in level is attributed to the collapse of voids in the tube bundle following the decrease in nuclear power. Liquid mass redistributes in the steam generator as more liquid within the downcomer is used to fill the space previously occupied by steam voids in the bundle. This mass redistribution results in a loss of water mass in the downcomer and, as a consequence, the narrow range level drops. Eventually, elevation heads in the bundle and downcomer regions equalize, mass redistribution decreases and excess feedwater results in an increase in water level.

For the turbine trip transient, the SG level response predicted by TRANFLOW compares favorably to that predicted by KSR010. TRANFLOW under predicts steam pressure due to differences in heat transfer correlations, but this has no discernable effect on the narrow range level response. This comparison demonstrates that TRANFLOW is fully capable of replicating the physics of level behavior during the turbine trip transient, i.e., TRANFLOW is fully capable of modeling the complex interrelationships between the collapse of voids in the tube bundle, liquid mass redistribution from downcomer to bundle, and interplay of elevation heads in the bundle and downcomer regions.

Figure 3.6-1: KSR010 Simulation of Turbine Trip Transient

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Figure 3.6-2: TRANFLOW and KSR010 Comparisons of SG Water Level and Steam Pressure Responses during Turbine Trip Transient

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4. CONCLUSION

TRANFLOW uses a variety of well-known mathematical methods and empirical correlations in order to provide accurate solutions to thermal-hydraulic design problems using standard steam generator parameters and assumptions. TRANFLOW has been extensively validated and qualified by a variety of sources and methods. It has been shown that the results produced using TRANFLOW are acceptable when making engineering justifications for the design of the steam generator components.

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A-12 Westinghouse Document CN-ICAT-95-001, Rev. 0, “Validation of ACSL Overall Plant Model (KSR010),” March 1995.

Appendix B

Excerpts from CENC-1246 (Analytical Report for Louisiana Waterford Unit Number 3 Steam Generator) Pertaining to the Original Hand Calculations that were Used Instead of CEFLASH-4A for the OSGs

Westinghouse Proprietary Class 2

Westinghouse

Westinghouse Electric Company
Nuclear Services
Waltz Mill Service Center
P.O. Box 158
Madison, Pennsylvania 15663
USA

Mr. Larry Rushing
Manager, Strategic Capital Projects
WF3 SG/RVCH Replacement Project
Entergy Operations Inc.
Arkansas Nuclear One
1448 S. R. 333
Russellville, AR 72802

Direct tel: 724 722-6078
Direct fax: 724 722-5412
e-mail: testada@westinghouse.com
Response Needed:
Your ref:
Our ref: LTR-WEN-RSG-08-221
CWTR3-08-291

June 4, 2008

SUBJECT: OSG Informational Request: CENC-1246, Rev. 0, and CENC-1512, Rev. 0
Project Licensing Schedule References: N/A
Waterford 3 Replacement Steam Generators Project

Dear Mr. Rushing,

Per Entergy's request to our Mr. Mehran Golbabai, we are transmitting the follow OSG Informational Reports:

- CENC-1246, Rev. 0: Analytical Report for Louisiana Waterford Unit Number 3 Steam Generator
- CENC-1512, Rev. 0: Addendum to the Analytical Report for Louisiana Waterford Unit No. 3 Steam Generator

For your convenience, these documents are linked with their entries in the Document Library. They also can also be found in the Waterford 3 eRoom in folder, "NSSS Licensing," and subfolder, "Miscellaneous."

As with all Westinghouse analytical reports, the referenced reports are the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. They are transmitted to you in confidence and trust, and you agree to treat these reports in strict accordance with the terms and conditions of the agreement under which they was provided to you.

Please feel free to contact Mr. Golbabai by phone at (860) 731-6444 or email at mehran.golbabai@us.westinghouse.com.

If you have any other questions, please do not hesitate to contact me by phone at (724) 722-6078 or email at testada@westinghouse.com.

Sincerely,

Damian A. Testa, Project Manager
Waterford 3 RSG Project

DAT/MG/las

Electronically Approved Records are Authenticated in the Electronic Document Management System

Links: CENC-1246, Rev. 0
CENC-1512, Rev. 0

cc: J. Holman, Entergy
M. Huff, Entergy
L. Humphrey, Entergy
B. Matthew, Entergy
M. Pappur, Entergy
J. Reese, Entergy
J. Taylor-Brown, Entergy
P. Albamonti, Westinghouse
D. Bersi, Westinghouse
M. Golbabai, Westinghouse
D. Heyer, Westinghouse
T. Miller, Westinghouse
J. Young, Westinghouse

PROPRIETARY INFORMATION

REPORT NUMBER CENC-1246
SUBJECT CATEGORY:
"ANALYTICAL REPORT"

COCC # 21848

PROPRIETARY INFORMATION

COMBUSTION ENGINEERING, INC.
NUCLEAR POWER SYSTEMS
COMPONENT ENGINEERING
C-E CONTRACT NO. 74270

ANALYTICAL REPORT FOR LOUISIANA WATERFORD UNIT NO. 3 STEAM GENERATOR

L. M. BARGER
L. D. CLIFT
J. C. LOWRY
D. G. SLACK

SEPTEMBER 1975

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Appendix C

Customer Comment Resolution

LTR-SGMP-17-107 – Model Notebook
Document Review Tracking Form

Document Number	LTR-SGMP-17-107,
Document Title	Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators
Revision Level	Rev. 0-A
Author of Document	Jivan Thakkar
Author Contact Info	423-697-5091
Date WEC Submitted Document	12/14/2017
Open Item(s)?	☒ Yes ☐ No If yes, see Document Tracking Log for Closure Status.

Entergy Reviewer(s)	James Hoss & Nick Petit
Entergy Reviewer Contact Info	
Disposition of Document*	
Date Entergy Reviewed Document	1/15/2018

***Disposition Key**

Approved:
Proceed to Formal Submission.
No changes necessary.
Downstream activity can proceed.

Approved as Noted:
Proceed to Formal Submission with noted changes. Downstream activity can proceed.

Revise & Resubmit:
Comments to be resolved and another draft submitted.
Downstream activities cannot proceed.

Item #	Page/Sec.#	Entergy Reviewer Comment	WEC Comment Resolution	Customer Reviewer Resolution Response
1.		Please provide a copy of the original hand calculations that were used instead of CEFLASH for the OSGs. This will allow us to withstand regulatory scrutiny.	WEC will provide the extracted pages from the previously transmitted Waterford Unit 3 AORs pertaining to the original hand calculations that were used instead of CEFLASH-4A for the OSGs. Please review the attached file, ***Entergy_Com1_Waterford3_OSG_SLB_AOR.pdf*** and let us know if this information meets your needs or if you have any questions. After the comment resolution, the information in the file will be included as an appendix to LTR-SGMP-17-107.	Concur with response

LTR-SGMP-17-107 – Model Notebook
Document Review Tracking Form

Item #	Page/Sec.#	Entergy Reviewer Comment	WEC Comment Resolution	Customer Reviewer Resolution Response
2.	11	As presented in Table 1 of this report, for the Waterford Unit 3 RSGs, the flow loads with the OSG design methodology (16 times steady state ΔPs) is higher than TRANFLOW calculated ΔPs for all SG internal components except for the secondary separators (steam dryers). Note that dryers are located nearest to the break location (steam outlet nozzle) and therefore experience initial and more rapid depressurization. As discussed in Section II of this report, the secondary separators have significant margins to the ASME code allowable. Also note that the steam dryers do not constitute a primary fluid (reactor coolant) pressure boundary. In the no Significant Hazards Analysis we will need to explain why that is not taking away margin. I realize that one analysis was done on the OSG and Tranflow was used in the RSG but it is still a change in analysis.	The Waterford Unit 3 OSGs were designed to satisfy all requirements of the 1971 ASME Code and the project/design specifications. In the early days of the nuclear industry, all nuclear components were designed conservatively with inherently higher margins through the use of manual calculations. The Waterford Unit 3 RSGs were also designed to satisfy all requirements of the 1998 ASME Code and the project/design specifications. With operating experience and availability of the advanced computer codes and computers, the nuclear components continue to be designed conservatively with margins. Please see the response to Question 3 beginning on page 12 of 14 of EVAL-17-30 SHC, Rev. 0-A for a discussion of why the use of the computer code TRANFLOW to calculate pressure differentials across the tube support plates (TSPs) during a postulated steam line break is acceptable. In summary, the use of TRANFLOW to calculate steam line break (SLB) pressure differentials across the TSPs in the RSGs is compared to NRC approved codes CEFLASH-4A, CEFLASH-4B, and RELAP5 and is determined to be technically equivalent. As the NEI 97-06, Rev. 3 steam generator performance criteria have been shown to be met during a postulated SLB using loads developed through the use of TRANFLOW, it is concluded that its use does not result in a significant reduction in margin of safety.	Concur with response
3.	Section II	Last statement of section II states that "The loads on SG internals for the post Waterford Unit 3 OSGs are calculated with the transient analysis codes CEFLASH-4A, CEFLASH-4B and	The statement may be revised as follows: "The loads on SG internals for the CE designed steam generators, subsequent to the Waterford Unit 3 OSGs,	Concur with response

LTR-SGMP-17-107 – Model Notebook
Document Review Tracking Form

Item #	Page/Sec.#	Entergy Reviewer Comment	WEC Comment Resolution	Customer Reviewer Resolution Response
		TRANFLOW." This statements seams out of place as the section was describing the OSGs and the previous sentence states that CFLASH was not used. Also, is this statement referring to WF3's RSG or other CE designed SGs.	are calculated with the transient analysis codes CEFLASH-4A (Reference 12) and CEFLASH-4B (References 13). Westinghouse uses the TRANFLOW code (Reference 14) for calculating the loads on the SG internals during the SLB and FWLB transients." Optionally the statement may be deleted from LTR-SGMP-17-107.	
4.				
5.				
6.				
7.				

Enclosure Attachment 5 to

W3F1-2018-0031

Westinghouse Letter CAW-18-4712,
Affidavit for LTR-SGMP-17-107 P-Attachment
Application for Withholding Proprietary Information
from Public Disclosure

Attachment contains 7 pages

As Attachment 6 contains information proprietary to Westinghouse Electric Company LLC, it is supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations.

Westinghouse Electric Company
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Cranberry Township, Pennsylvania 16066
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Rockville, MD 20852

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Direct fax: (724) 940-8542
e-mail: greshaja@westinghouse.com

CAW-18-4712

February 26, 2018

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators (Proprietary)

The Application for Withholding Proprietary Information from Public Disclosure is submitted by Westinghouse Electric Company LLC ("Westinghouse"), pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-18-4712 signed by the owner of the proprietary information, Westinghouse. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Entergy Operations, Inc.

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-18-4712, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 2 Suite 259, Cranberry Township, Pennsylvania 16066.

J A Gresham

James A. Gresham, Manager
Regulatory Compliance

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC ("Westinghouse") and declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

Executed on: 2/26/18

James A. Gresham, Manager
Regulatory Compliance

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

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Nuclear Regulatory Commission's ("Commission's") regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

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

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, is to be received in confidence by the Commission.

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

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-SGMP-17-107 P-Attachment, "Acceptability of the TRANFLOW Computer Code for Steam Line Break Internal Pressure Loads for the Waterford Unit 3 Replacement Steam Generators" (Proprietary), for submittal to the Commission, being transmitted by Entergy Operations letter. The proprietary information as submitted by Westinghouse for use by Entergy Operations, Inc., for Waterford Unit 3 demonstrates the acceptability of using the computer code, TRANFLOW, to calculate replacement steam generator secondary side internal loads during a postulated steam line break event.

(a) This information is part of that which will enable Westinghouse to describe the computer code, TRANFLOW, and to benchmark calculated pressure drops using TRANFLOW with other NRC approved computer code results.

(b) Further, this information has substantial commercial value as follows:

(i) Westinghouse can sell the use of similar information to its customers for the purpose of meeting NRC requirements for licensing documentation supporting the use of the computer code, TRANFLOW, during replacement steam generator design.

(ii) Westinghouse can sell support and defense of this information to customers, if the need arises.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which is utilized by Westinghouse for replacement steam generator design.

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

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

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

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in support of a license amendment to obtain approval to use the computer code, TRANFLOW, for calculating secondary side internal loads during a postulated steam line break event in the Waterford Unit 3 replacement steam generators, and may be used only for that purpose.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

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