CENPD-289-NP
Supplement 1
Revision 0

August 2022

Use of Inert Replacement Rods in CE 16x16 Next Generation Fuel (CE16NGF™)

Westinghouse

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August 2022

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Table of Contents

1.0 INTRODUCTION AND BACKGROUND 1
1.1 INTRODUCTION 1
1.2 BACKGROUND 1
2.0 EFFECT OF ROD REPLACEMENT ON NEUTRONIC PERFORMANCE 4
3.0 EFFECT OF ROD REPLACEMENT ON THERMAL HYDRAULIC PERFORMANCE 8
4.0 EFFECT OF ROD REPLACEMENT ON MECHANICAL PERFORMANCE 10
5.0 EFFECT OF ROD REPLACEMENT ON PLANT SAFETY ANALYSIS 11
6.0 CONCLUSIONS 14
7.0 REFERENCES 15
APPENDIX A SPECIAL CHF TESTS WITH INERT RODS 16
APPENDIX B SPECIAL CHF TESTS WITH INERT RODS – DATA SUMMARY 26
APPENDIX C SPECIAL TESTS - ABB-NV AND WLOP STATISTICAL OUTPUT 31
GLOSSARY 36

1.0 Introduction and Background

1.1 Introduction

The purpose of this supplement is to provide additional justification on the applicability of current Westinghouse methodology for analysis of Combustion Engineering 16x16 Next Generation Fuel (**CE16NGF™**[1]) configurations containing inert stainless steel replacement rods. Replacement of fuel rods with the inert rods in a fuel assembly is also referred to as fuel reconstitution. Note that the applicability of Westinghouse methodology for "Class A" type configurations in **CE16NGF** fuel was already implicitly approved in Reference 2. The purpose of this supplement to CENPD-289-P (Reference 1) is to provide justification for the application of the approved **CE16NGF** methodology (Reference 2) for inert rod configurations beyond the original Class A configurations and subject to the limitations provided in Section 6.

This supplement has been prepared using the licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP), Reference 9. The design objectives of the fuel assemblies containing the inert rods are the same as those described in Section 4.2 of the SRP, i.e. to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, 2) fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) coolability is always maintained.

1.2 Background

Reference 1 describes the methodology for analysis of reconstituted fuel assemblies containing inert replacement rods for Combustion Engineering Nuclear Steam Supply System (CE-NSSS) designed PWRs. The topical report was submitted in response to GL 90-02 which required NRC approval for methods used to analyze core configurations containing "inert" rods. Conservatism of DNB methodology was a primary concern of the NRC. Similar methodology submittals were made by the other fuel vendors (including Westinghouse in Reference 5).

Reference 1 defines two "classes" of inert rod configurations. The "Class A" Configurations are defined as:

[ ] a,c

[1] CE16NGF is a trademark or registered trademark of Westinghouse Electric Company LLC, its affiliates and/or its subsidiaries in the United States of America and may be registered in other countries throughout the world. All rights reserved. Unauthorized use is strictly prohibited. Other names may be trademarks of their respective owners.

Non-Class A configurations are defined as all configurations that do not satisfy Class A definitions.

In Reference 1, Class A was defined as those configurations of inert rods within an assembly that had been demonstrated by analysis and DNB test data to result in no significant impact on fuel performance or plant safety analysis provided that total number of inert rods in the core is within assumptions of safety analysis. Reference 1 also provided data to demonstrate that although non-Class A configurations could result in non-negligible impact, the standard approved analysis methods could be used to perform explicit cycle specific analysis to determine and accommodate impact.

However unresolved Requests for Additional Information (RAIs) from the NRC related to application of the Critical Heat Flux (CHF) test data to configurations involving large numbers of inert rods resulted in the NRC Safety Evaluation Report (SER) giving approval to only Class A configurations. The specific concern was the lack of CHF test data for non-Class A configurations. In order to obtain timely approval to support upcoming reloads, an SER restriction was accepted in 1999 that limited approval to only the Class A configurations. The SER restriction was not a major concern at the time since rod swaps could be performed to meet Class A restrictions in most cases.

Since the initial approval of Reference 1 in 1999, the following changes have been implemented:

- A newer NRC-approved fuel assembly design, **CE16NGF** (Reference 2) has been implemented on several CE-NSSS type plants. This new fuel assembly design, which has mixing vanes (MV) on the mid grids and Intermediate Flow Mixer (IFM) grids required new CHF tests to support the development of a new CHF correlation, WSSV/WSSV-T (Reference 4). The **CE16NGF** fuel design with the Mixing Vanes and IFM grids provides significant CHF margin improvements over the original Combustion Engineering 16x16 Standard (CE16STD) fuel design. Reference 2 confirmed the applicability of Reference 1 to the **CE16NGF** design for Class A configurations.
- The implementation of **CE16NGF** has eliminated the primary cause of fuel failure **[** **]**[a,c] to meet the SER conditions and limitations in Reference 1. With the **CE16NGF** design the current impetus for use of inert replacement rods is to prevent fuel failure **[** **]**[a,c]
- The development and implementation of a new CHF correlation for non-vaned CE16x16 fuel or the non-vaned axial region of **CE16NGF** design, ABB-NV (Reference 3), which included CHF tests not used for the original CE-1 CHF correlation.
- The acquisition of ABB Nuclear by Westinghouse has enabled integration of NRC-approved Westinghouse methodologies applicable to CE-NSSS plants (e.g., VIPRE-W thermal-hydraulic code, WLOP CHF correlation, Westinghouse Thermal Design Procedure, etc). The **CE16NGF** fuel design is similar to the Westinghouse-NSSS fuel

assembly designs containing mixing vane grid spacers and IFM grids described in the Westinghouse Fuel Reconstitution topical report (Reference 5). This approved topical report has justified the conservatism of the Westinghouse DNB methodology for inert rod configurations that are well beyond the Class A type configurations defined in Reference 1.

The changes listed above justify the basis for relaxing the SER conditions and limitations on the original methodology in CENPD-289-P/NP-A as they apply to the **CE16NGF** design. The appropriate SER conditions and limitations would have direct benefits on plant operations, safety, and fuel cost.

[ ] **a,c**

Based on these considerations this supplement is being provided to justify application of the current **CE16NGF** approved methodology (Reference 2) to non-Class A inert rod configurations in **CE16NGF** fuel. This methodology will be applied for the cycle specific analysis of all non-Class A inert rod configurations to determine if additional **[**

]a,c

assure conservative operation within the allowed licensed limits.

Note that Reference 1 has already concluded that most parameters and analyses for the CE16STD assembly are not affected by replacement of fuel rods with inert stainless steels rods. These conclusions continue to hold for **CE16NGF** assemblies. This supplement only addresses those items which were identified in the Reference 1 SER (specifically DNB CHF and Fuel Mechanical Integrity) as being unresolved with regard to the non-Class A type configurations.

2.0 Effect of Rod Replacement on Neutronic Performance

Reference 1 addressed the basic neutronics effects of the replacement of depleted fuel rods with stainless steel insert or replacement rods as the inert rods. This reference showed that the primary effects of the inert rod replacement is on the local power peaking in fuel rods adjacent to the inert rod and on the incore instrument response for situations where the inert rod is placed near the center instrument tube. These effects are the result of an increase in the thermal neutron flux at and in the vicinity of the stainless steel rod. This increase in thermal flux is due to the fact that the thermal neutron absorption cross section of stainless steel is significantly less that the uranium rod that it replaces. The increase in thermal neutron flux results in a moderate increase in the power of fuel rods in the vicinity of the stainless steel rod. If the inert rod replacement is adjacent to an instrumented guide tube, a small increase in the instrument flux will also occur which, if uncorrected, will result in a slightly over conservative measurement of the assembly power and burnup. However, the effects are small for most configurations of interest and in fact can be neglected for the Class A type configurations.

The approved neutronics design methodology (Reference 2) is used to explicitly model the effects of the non-Class A type configurations. The configuration of inert rods within the assembly fuel assembly is explicitly modeled in the fuel assembly lattice code. The results of the lattice code are input to the core design code to determine the impact on core wide power distribution and instrument responses. Benchmarks of the lattice code to critical experiments and Monte Carlo calculations containing a limited number of inert rods has shown no significant degradation in accuracy. A constraint (#3 of Section 6) has been added to the Limits of Applicability to assure that large clusters of inert rods that might degrade the accuracy of the predictions cannot occur.

The neutronic impact of a variety of Class A and non-Class A inert rod replacement configurations in **CE16NGF** fuel was investigated using the methodology described in Reference 2 (which is also consistent with those used in Reference 5 for analysis of inert rod configurations in fuel designs for Westinghouse type PWRs). These calculations (See Figure 2-1 and Figure 2-2 for examples of anticipated typical non-Class A inert rod configurations) show power peaking and instrument flux impacts for a **CE16NGF** assembly that are very similar to those reported in Reference 1 for CE16STD fuel with a fuel rod outside diameter of 0.382 inches. Although the local fuel rod power increases for these non-Class A configurations may not be negligible, they would be explicitly considered in the cycle specific reload safety analysis and incorporated, if necessary, into the cycle specific core monitoring and trip setpoints.

Based on these considerations it is concluded that the neutronics methods of Reference 2 is applicable for analysis of both Class A and non-Class A type inert rod configurations in **CE16NGF** fuel.

Figure 2-1
Fuel Rod Power Increase for Typical Non-Class A Inert Rod Configurations

a,c

Figure 2-1 (continued)
Fuel Rod Power Increase for Typical Non-Class A Inert Rod Configurations

a,c

[ ]

Figure 2-2
Assembly Power to Incore Rhodium Instrument Flux Ratio for Typical Non-Class A Inert Rod Configurations

a,c

3.0 Effect of Rod Replacement on Thermal Hydraulic Performance

Reference 1 addressed the basic thermal hydraulic effects of the replacement of depleted fuel rods with stainless steel insert rods. This reference showed that the CE-1 CHF correlation remained a conservative predictor of critical heat flux for all Class A configurations. Reference 1 also presented justification for why the CE-1 CHF correlation would remain conservative for most non-Class A configurations. However, the SER on Reference 1 limited applicability to Class A configurations due to the limited CHF test data for non-Class A configurations.

The implementation of **CE16NGF** design has significantly positive impacts on the thermal hydraulic performance of the fuel assembly. In addition to the fuel rod diameter change from 0.382 to 0.374 inches, the two major changes (See Reference 2) relative to thermal hydraulics are:

- The addition of mixing vanes on six or seven mid-grids
- The addition of two Intermediate Flow Mixing (IFM) grids with mixing vanes to the assembly.

The mixing vane and IFM features are also present on the Westinghouse fuel designs for which Reference 5 has confirmed the applicability of the standard approved methodology for inert rod configurations.

The impacts of the mixing vanes and the IFMs are to redistribute the flow within the channels and assembly so that "cold wall" effects on the CHF is significantly reduced. Even without the mixing vanes the structural grid spacers near the inlet region of the **CE16NGF** assembly also enhance flow mixing as reflected in the ABB-NV correlation. The existing DNB correlations and the approved DNBR limits continue to be applied to the **CE16NGF** configurations containing the inert rods presented in Section 2. Values of the channel multipliers (also referred to as cold wall multipliers) for ABB-NV, WSSV/WSSV-T and WLOP correlations for several subchannel types of interest are shown in Table 3-1. As shown in Appendix A, the CHF for the special test with an inert rod adjacent to the guide thimble was conservatively predicted with the ABB-NV and WLOP correlations with the correlation cold wall ratio (or the channel heated diameter ratio), Dhm/Dh, limit of **[ ]**[a,c] so no additional conservatism is required. To assure that the conclusions from the test results remain applicable, a constraint (#2 of Section 6) has been added to the Limits of Applicability.

Table 3-1

Typical Channel Multipliers for DNB Correlations

a,c

4.0 Effect of Rod Replacement on Mechanical Performance

Reference 1 addressed the basic mechanical performance effects of the replacement of depleted fuel rods with stainless steel insert rods. This reference showed that the primary effect of the inert rod replacement is a decrease in the weight of the fuel assembly. The minimum fuel assembly weight is an important consideration for the confirmation of adequate fuel assembly holddown to prevent uplift from the core support structure due to core flow and also the frequency response of the fuel assembly during seismic events.

The confirmation of adequate fuel assembly holddown is a plant specific calculation that considers the minimum fuel assembly weight used in the plant specific analyses of record (AOR). The implementation of **CE16NGF** design has decreased the weight of the fuel assembly relative to CE16STD fuel. This decrease is the result of a small decrease in the fuel rod diameter and also because of the use of annular fuel pellets in the axial blanket regions of some fuel rods. These changes necessitated an increase in the holddown spring force for the **CE16NGF** fuel assembly (see Reference 2). The **CE16NGF** fuel assembly holddown AOR consider a range of assembly weights to accommodate the potential for a large number of annular fuel pellets in the fuel rods.

The replacement of fuel rods with inert stainless steel rods will further decrease the weight of the fuel assembly. The weight of a **CE16NGF** fuel rod is approximately **[ ]**[a,c] lbs while the weight of a **CE16NGF** inert stainless steel rod is approximately **[ ]**[a,c] lbs. The maximum number of inert stainless steel rods that could be inserted into a **CE16NGF** fuel assembly while retaining positive fuel assembly holddown margin, accounting for the maximum number of annular pellets in the assembly, has been shown to be bounded by the current AOR and will be checked each cycle.

With regard to response during seismic LOCA, it has been confirmed that the insertion of up to **[ ]**[a,c] stainless steel rods in the **CE16NGF** assembly will not invalidate the assumptions used in any of the plant specific seismic LOCA load calculations. The existing AOR for fuel assembly mechanical integrity during seismic LOCA events remains applicable for up to **[ ]**[a,c] replacement stainless steel rods in pre-analyzed configurations in a **CE16NGF** fuel assembly. The reason is that for this type of fuel assembly the significant assembly parameters of interest (weight, stiffness, and frequency), are compatible with those for a standard **CE16NGF** assembly with no stainless steel rods. The fuel assembly spacer grid strength and stiffness are not affected by the substitution of stainless steel rods for the fuel rods. The reduction in assembly weight is a small percentage of the total fuel assembly weight (< 0.5%) and does not have a significant effect on the dynamic behavior of the fuel assembly. Likewise, for the pre-analyzed configurations of **[ ]**[a,c] SS rods there is no significant change in assembly stiffness or frequency. Therefore, the assembly and rod force margins from the seismic and LOCA excitations will also be bounded by, or similar to, the existing AOR. Some configurations of inert rods, such as those having a large number of inert rods grouped together on the periphery of the fuel assembly, may require additional plant specific assessments using methods consistent with the AOR.

In summary the existing plant specific AOR for fuel assembly mechanical integrity during seismic and LOCA events will remain applicable for up to **[ ]**[a,c] inert replacement rods.

5.0 Effect of Rod Replacement on Plant Safety Analysis

Reference 1 addressed the effects of the replacement of depleted fuel rods with stainless steel insert rods on the plant safety analysis. This reference showed that the primary effects of the inert rod replacement is on the local power peaking in fuel rods adjacent to the inert rod and on the core average linear heat rate (kW/ft). However, the effects are small for most configurations of interest and in fact can be neglected for the Class A type configurations.

The replacement of fuel rods with inert rods will result in an increase in the core average power density (kW/ft) which is used in several safety analyses to determine the fuel rod power associated with a given peaking factor. The core average power density used in the plant specific safety analysis typically assumes a small number of non-fuel rods (usually somewhere around **[** **]**[a,c] inert rods per core). It is anticipated that in most actual cores there will be only a few stainless steel rods in any one assembly and very few total in the core. A maximum limit of **[** **]**[a,c] has been imposed to insure that sufficient margins to regulatory limits will be maintained.

Current reload safety analysis procedures require that the actual number of non-fuel rods in the core be confirmed each cycle to be less that the number assumed in the plant safety analysis. If it is determined that the value exceeds this, then the **[** **]**[a,c]

The methodology to perform DNB calculations for **CE16NGF** design is described in Reference 2 which is supplemented by References 6, 7 and 8. The impact of inert rod replacement on DNB due to increase in pin power of fuel rods within the reconstituted fuel assembly and in the surrounding assemblies is addressed by employing a subchannel code such as TORC and VIPRE-W. The DNBR distribution across the fuel bundle length is predicted through the applicable CHF correlations for the fuel type. The DNBR calculations encompass the setpoint wide range of operating conditions such that the response of different conditions and peaking factor changes on the minimum DNBR is examined. **[** **]**[a,c]

While the low-powered and high-burned assemblies are typical candidates for inert rod replacement, a possibility may exist to require inert rod replacement in once burned assemblies which are of relatively high power. Even with the low-powered inert rod assemblies, the impact on power peaking can also be carried over to the neighboring rods in surrounding assemblies. **[** **]**[a,c]

For CE-NSSS plants that are equipped with COLSS and CPC online systems, the inert rod replacement is to be evaluated on a case-by-case basis using the approved codes and CHF

correlations. **[**

][a,c]

For the transient events, such as sheared shaft/seized rotor, where the statepoint conditions fall outside of the setpoint wide range of operating conditions, additional evaluation is performed for the loading pattern containing inert rods. Since the DNB results of such transient events are not incorporated into COLSS/CPC online systems, **[**

][a,c]

For an illustration, the impact of a variety of Class A and non-Class A inert rod replacements, as shown in Figure 2-1, on DNB in **CE16NGF** fuel was evaluated using the methodology described in Reference 2. The same analytical steps as described above were followed for each of the inert rod configurations. The DNBR distribution across the **CE16NGF** bundle length was predicted through WSSV/WSSV-T and ABB-NV DNB correlations. **[**

][a,c] presented in Table 5-1.

Table 5-1

a,c

6.0 Conclusions

Based on the discussions presented here-in it is concluded that the methodology described in Reference 2, including the DNB methodology in Reference 2 supplemented with the approved DNB correlations and evaluation methods in References 6, 7 and 8, is applicable for analysis of inert rod configuration in **CE16NGF** type assemblies subject to the following limitations:

[ ] a,c

The evaluation method described in this report in supplement to CENPD-289-P-A will be applied for the cycle specific analysis of all **CE16NGF** non-Class A inert rod configurations within the above limitations to **[**

][a,c] The conclusions regarding acceptability of Class A inert rod configurations previously documented in CENPD-289-P-A continue to apply. **[**

][a,c]

7.0 References

1. CENPD-289-P-A, "Use of Inert Replacement Rods in ABB CENF Fuel Assemblies," July 1999.
2. WCAP-16500-P-A, "CE 16x16 Next Generation Fuel Core Reference Report," August 2007.
3. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.
4. WCAP-16523-P-A, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," August 2007.
5. WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," July 1993.
6. WCAP-14565-P-A, Addendum 1-A, "Addendum 1 to WCAP-14565-P-A, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," June 2002.
7. WCAP-14565-P-A, Addendum 2-P-A, "Addendum 2 to WCAP-14565-P-A, Extended Application of ABB-NV Correlation and Modified ABB-NV Correlation WLOP for PWR Low Pressure Applications," April 2008.
8. WCAP-18240-P-A, "Westinghouse Thermal Design Procedure (WTDP)," April 2020.
9. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 4.2 – Fuel System Design, Revision 3, March 2007.

Appendix A Special CHF Tests with Inert Rods

To confirm the conservative application of the ABB-NV and WLOP correlations for replacement inert rods for both the Class A configurations, defined in Reference 1, and non-Class A configurations, data from three (3) special tests have been examined with the VIPRE-W code. Based upon available test data, comparisons are made between reference tests (tests 47 and 52 where all test rods are heated) and special tests with inert (non-heated) rods for one Class A configuration (test 73) and one non-Class A configuration (test 70). A third special test (test 72), which simulates the corner of four assemblies with perimeter straps is also examined with VIPRE-W. It is noted that the Class A configuration special test (73) is the same test documented in Appendix B of Reference 1 except the ABB-NV correlation is applied in place of the CE-1 correlation. The third special test (72) is documented in Reference 3 with the TORC thermal-hydraulic code where the ratio of the measured to ABB-NV predicted heat flux is consistent with the results with the VIPREW code shown below. The non-Class A configuration special test (70) had an inert rod adjacent to the large guide thimble and provides additional confirmation that the non-Class A configuration is conservatively analyzed with the ABB-NV and WLOP correlations. The reference tests (47 and 52) were included in the ABB-NV correlation database, documented in Reference 3. All tests have a uniform axial power shape. The radial geometries are presented in Figures A-1 through A-5.

For ABB-NV, the impact of the inert rod in the special tests is assessed by comparison with a reference test. These comparisons are performed based on the raw data in the form of hot rod heat flux versus inlet temperature and based upon the measured to ABB-NV predicted (M/P) ratio at the MDNBR local conditions. For the raw data comparisons, the data are sorted by pressure and flow to generate plots. Typical comparison plots of the hot rod heat flux versus inlet temperature for three nominal flows are shown in Figures A-6 and A-7. The tests with inert rod(s) (70 and 73) had a higher test section and hot rod heat flux than the reference tests (52 and 47, respectively) without inert rods. Furthermore, the test section power needed to reach DNB was higher for the special tests in addition to the higher heat flux due to the reduced number of test rods.

In addition to the raw data analysis, a subchannel analysis is performed with the VIPRE-W code and the ABB-NV correlation. Since Test 70 had multiple points at low pressure, the data are also examined with the VIPRE-W code and the WLOP correlation. The ABB-NV correlation, References 3 and 6, was developed with eleven uniform axial power tests and three non-uniform axial power tests shown in Table 6.2-2 in Reference 3. These tests include the reference tests 47 and 52. The test simulating the corner of four assemblies, Test 72, was identified as a special test with large M/P in Section 3.4 of Reference 3. The results for Test 73 provided in Reference 3 are based on setting the matrix heated hydraulic diameter, Dhm, to [ ]^{a,c} since that is the matrix heated hydraulic diameter for that test. However, for reactor Class A cases, the typical matrix heated hydraulic diameter would remain as **[ ]^{a,c}** so that is used for the evaluation of Test 73 data in this supplement. [

]^{a,c}

For the subchannel analysis with the ABB-NV correlation, the data for the test simulating the corner of four assemblies, Test 72, are run with an assumed inert rod on a side channel and an inert rod on the corner channel. Based on the higher heat flux for special tests with inert rods shown in Figures A-6 and A-7, the test section average heat flux with assumed inert rods is conservatively assumed to be the same as the Test 72 results.

The M/P ratio for the tests with the ABB-NV correlation is provided below when the value of Dhm is based on the nominal reference test section matrix heated hydraulic diameter of [ ]a,c inches.

[ ] a,c

* Rod 20 assumed to be inert

** Rod 22 assumed to be inert

NTS – Test section number

NP – Number of data points

AVG – M/P mean

SDF – Standard deviation of M/P distribution.

The M/P ratio for the special test 70, the only special test with data at low (< 1500 psia) pressure, with the WLOP correlation is provided below.

[ ] a,c

Based upon this result, the WLOP correlation remains applicable to analyze inert replacement rods. Test 47 was the only reference test with low pressure points and from Reference 7, the mean for 50 points was **[**

]a,c

[

]a,c

This confirms the ABB-NV and WLOP cold wall term is conservative for application with inert rods, and there is no change in approved ABB-NV and WLOP correlations needed to account for additional cold wall effects due to inert rods.

As stated above, since the special tests with inert rods (70 and 73) had higher measured hot rod heat flux compared to the reference tests (Tests 52 and 47), the test section heat flux for the bundle periphery test is conservatively assumed to be unchanged for the cases with an assumed inert test rod. This reduces the measured to predicted ratio since the predicted CHF value is higher due to lower local quality and higher mass velocity for the case with an assumed inert rod. However, the M/P value remains well above 1.0 plus correlation standard deviation showing the ABB-NV correlation is conservative for the peripheral channels.

Based upon the very conservative ABB-NV M/P results for the special tests relative to the reference tests, one can conclude that the existing NRC-approved DNB correlations such as ABB-NV can be used for evaluating impact of the fuel rod radial peaking factor and heated hydraulic diameter changes due to the inert rod(s) on DNBR margin. Based on the results from the special tests, the existing DNB correlation with the TORC/VIPRE-W subchannel modeling approach in this calculation is conservative for evaluation of both Class A and non-Class A configurations of reconstituted fuel assemblies.

Figure A-1
Radial Geometry for Special Class A CHF Test Section 73
Heated Length, 12.5 feet
Grid Spacing, 15.7 inches

a,c

Figure A-2
Radial Geometry for Reference CHF Test Section 47
Heated Length, 12.5 feet
Grid Spacing, 14.3 inches

a,c

Figure A-3
Radial Geometry for Special Non-Class A CHF Test Section 70
Heated Length, 7.0 feet
Grid Spacing, 14.3 inches

a,c

Figure A-4
Radial Geometry for Reference CHF Test Section 52
Heated Length, 7.0 feet
Grid Spacing, 14.3 inches

a,c

Figure A-5
Radial Geometry for Special CHF Test Section 72
Heated Length, 7.0 feet
Grid Spacing, 14.3 inches

a,c

Figure A-6
Raw Hot Rod Heat Flux vs. Inlet Temperature Data
Reference Test 47 and Special Test 73

a,c

Figure A-7
Raw Hot Rod Heat Flux vs. Inlet Temperature Data
Reference Test 52 and Special Test 70

a,c

Appendix B Special CHF Tests with Inert Rods – Data Summary

A detailed summary of the Test 70 and Test 72 special test ABB-NV database is shown in Table B-1. A detailed summary of the Test 70 WLOP database is shown in Table B-2 since the MDNBR channel for the two correlations is different for some test runs. The data for Tests 47, 52 and 73 are provided in Reference 3. The tables in this appendix summarize the raw data from data files of the test reports, the test geometry information needed for the correlation evaluation, the predicted local coolant conditions for the MDNBR channel and elevation taken from the VIPRE-W runs. The tabulation presented here gives the data from the special test CHF experiments within the ABB-NV and WLOP correlation pressure and flow parameter limits in References 3, 6 and 7. Nomenclature for heading abbreviations in Appendix B are defined below:

TS	=	Test Section Number
TD	=	Test Section Type (UM is Uniform Shape without Thimble, UT is Uniform Shape with Thimble, NM is Non-Uniform Shape without Thimble and NT is Non-Uniform Shape with Thimble)
Run	=	Run Number
Pr	=	Test Section Pressure (psia)
Tin	=	Test Section Inlet Temperature (°F)
Gavg	=	Average Test Section Mass Velocity (Mlbm/hr-ft^2)
Qavg	=	Test Section Critical Bundle Average Heat Flux (MBtu/hr-ft^2)
DROD	=	Primary DNB Rod Thermocouple Number
DCH	=	VIPREW Subchannel Number Where Local Coolant Conditions are extracted
GL	=	Local Mass Velocity in CHF Channel at CHF Site (Mlbm/hr-ft^2)
XL	=	Local Quality in CHF Channel at CHF Site
CHFM	=	Measured CHF (MBtu/hr-ft^2)
F_C	=	Non-uniform Shape Factor = 1.00 for Uniform Axial Power Shape Based on Optimized F_C for Non-Uniform Axial Power Shape
GS	=	Nominal Grid Spacing (in)
HL	=	Heated Length to CHF Site (in)
DG	=	Distance from Middle of Grid to CHF Site (in)
De	=	Wetted Hydraulic Diameter of CHF Channel (in)
Dh	=	Heated Hydraulic Diameter of CHF Channel (in)
Dhm	=	Heated Hydraulic Diameter of Matrix Channel (in)

Table B-1

Special Test ABB-NV Database for Tests 70 and 72

a,c

Table B-1 Continued

Special Test ABB-NV Database for Tests 70 and 72

a,c

Table B-1 Continued

Special Test ABB-NV Database for Tests 70 and 72

a,c

Table B-2

Special Test WLOP Database for Test 70

a,c

Appendix C Special Tests - ABB-NV and WLOP Statistical Output

A detailed summary of the statistical output for special tests 70 and 72 with the ABB-NV correlation is given in Table C-1. A detailed summary of the statistical output of special test 70 with the WLOP correlation is given in Table C-2. For each test run in Tables C-1 and C-2, the values for the correlation variables, the measured CHF and the ABB-NV or WLOP correlation predicted CHF are given, along with the value for the M/P CHF ratio. The individual test section statistics are given at the end of the output in Tables C-1 and C-2. Nomenclature for heading abbreviations in Appendix C is defined below:

TS	=	Test Section Number
TD	=	Test Section Type (UM is Uniform Shape without Guide Thimble, UT is Uniform Shape with Guide Thimble, NM is Non-Uniform Shape without Guide Thimble, NT is Non-Uniform Shape with Guide Thimble)
Press	=	Test Section Pressure (psia)
GL	=	Local Mass Velocity in CHF Channel (Mlbm/hr-ft^2)
XL	=	Local Quality in CHF Channel (fraction)
GS	=	Upstream Nominal Grid Spacing, Middle of Grid to Middle of Grid (in)
HL	=	Heated Length to CHF Site (in)
DG	=	Distance from Middle of Grid to CHF Site (in)
De	=	Wetted Hydraulic Diameter of CHF Channel (in)
Dh	=	Heated Hydraulic Diameter of CHF Channel (in)
Dhm	=	Heated Hydraulic Diameter of Matrix Channel (in)
CHFM	=	Measured CHF (MBtu/hr-ft^2)
CHFP	=	ABB-NV or WLOP Predicted CHF (MBtu/hr-ft^2)

Table C-1

Statistical output of ABB-NV Correlation

a,c

Table C-1 Continued

Statistical output of ABB-NV Correlation

a,c

Table C-1 Continued

Statistical output of ABB-NV Correlation

a,c

Table C-2

Statistical output of WLOP Correlation

a,c

GLOSSARY

Acronym/Abbreviation	Definition
ABB-NV	Critical heat flux correlation for non-vaned fuel
ANC	Westinghouse neutronics computer code
CE-1	Original CE CHF correlation for non-vaned fuel
CE16NGF™	Next Generation 16x16 Fuel for CE-NSSS plants
CE16STD	Standard (pre-NGF) 16x16 Fuel for CE-NSSS plants
CEA	Control Element Assembly
CE-NSSS	Combustion Engineering Nuclear Steam Supply System
CETOP-D	Thermal margin algorithm and computer code
CHF	Critical Heat Flux
COLSS	Core Operating Limit Supervisory System
CPC	Core Protection Calculator
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
IFM	Intermediate Flow Mixer
LHR	Linear Heat Rate (e.g. kw/ft)
LOCA	Loss of Coolant Accident
M/P	Measured CHF to Predicted CHF Ratio
MV	Mixing Vane
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
SER	Safety Evaluation Report
TORC, VIPRE-W	Detailed design core thermal hydraulic subchannel code
WLOP	CHF correlation for Low Pressure applications
WSSV, WSSV-T	NGF CHF correlations for side supported mixing vaned fuel