PRESSURIZED WATER REACTOR OWNERS GROUP

PWROG-16043-NP-A
Revision 2

WESTINGHOUSE NON-PROPRIETARY CLASS 3

PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09: "IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH" FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS

Analysis Committee

PA-ASC-1169, Revision 4

November 2019

Westinghouse

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PA-ASC-1169, Revision 4

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November 2019

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NRC FINAL SAFETY EVALUATION

This section contains the following documents:

1. NRC cover letter, "Final Safety Evaluation for Pressurized Water Reactor Owner's Group Topical Report PWROG-16043-P, Revision 2, "PWROG Program to Address U. S. Nuclear Regulatory Commission Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" (EPID: L-2018-TOP-0021), dated October 31, 2019.

2. "Final Safety Evaluation by the of Nuclear Reactor Regulation for Topical Report PWROG-16043-P, Revision 2, "PWROG Program to Address U. S. Nuclear Regulatory Commission Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" Pressurized Water Reactor Owners Group (PWROG), dated May 17, 2019.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 31, 2019

Mr. W Anthony Nowinowski
Executive Director
PWR Owners Group,
Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-16043-P, REVISION 2, "PWROG PROGRAM TO ADDRESS U S. NUCLEAR REGULATORY COMMISSION INFORMATION NOTICE 2012-09: 'IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH' FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS" (EPID: L-2018-TOP-0021)

Dear Mr. Nowinowski:

By letter dated February 1, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17039B050), the Pressurized Water Reactor (PWR) Owners Group (PWROG or the applicant), submitted to the U.S. Nuclear Regulatory Commission (NRC) staff for review licensing topical report (TR) PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE [Combustion Engineering] PWR [pressurized water reactor] Fuel Designs" ((ADAMS Package Accession No. ML17039B061), henceforth referred to as the TR). Subsequent letters dated March 27, 2018, May 15, 2018, and May 15, 2018 (ADAMS Accession Nos. ML18100A053, ML18143B462, and ML18144A760, respectively), provided additional information that supplemented the information provided in the February 1, 2017, submittal.

The NRC staff review determined that the information provided in the TR and responses to NRC staff requests for additional information adequately demonstrates that the proposed methodologies to address end-of-life (EOL) effects on spacer grids and to recover margin through credit for flowing water damping are acceptable for use, subject to the limitations and conditions contained in the enclosed draft safety evaluation (SE), with existing methodologies that the NRC has previously found to be acceptable for analysis of fuel assembly structural behavior during seismic and loss-of-coolant-accident events.

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W. Nowinowski - 2 -

By letter dated August 22, 2018 (ADAMS Accession No. ML18186A625), the NRC staff provided the draft SE to the PWROG for review and comment. Per email correspondence on January 16, 2019 (ADAMS Accession No. ML19203A418), the PWROG provided comments to the NRC staff. Per email correspondence on April 5, 2019, the NRC staff provided a revised draft SE to the PWROG for review and comment. Per email and its enclosure dated May 14, 2019, and May 16, 2019, the PWROG provided comments to the NRC staff. The NRC staff disposition tables for the draft SE comments can be found in ADAMS Accession Nos. ML19071A239 and ML19242B646, respectively.

In accordance with the guidance provided on the NRC website, we request that the PWROG publish an approved version of PWROG-16043-P, Revision 2 within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. For -NP versions, the PWROG shall strike the proprietary information markings in this letter and make the appropriate redactions and adjustments to document security classifications to the attached SE. Also, it must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved version shall include an "-A" (designating approved) following the TR identification symbol. As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an appendix to the accepted version.

2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, The PWROG will be expected to revise the TR appropriately or justify their continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify their continued applicability or evaluate their plant using the revised TR.

If you have any questions, please contact Jason Drake at 301-415-8378.

Sincerely,

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Project
Office of Nuclear Reactor Regulation

Docket No. 99902037

Enclosure:
Final SE (~~Proprietary~~)

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U.S. Nuclear Regulatory Commission
Comment Resolution Table for PWROG-16043

Comment Number	Text Location		PWROG Comment (paraphrased)	NRC Response
	Page	Line		
1	3	22-31	It is not clear what the purpose is to reference Appendix S of 10 CFR Part 50. Depending on the vintage of licensing for plants, Appendix S of 10 CFR Part 50 as well as Appendix A of 10 CFR Part 100 may not be the licensing basis.	The staff agrees that not all plants are licensed under Appendix S of 10 CFR Part 50. However, some plants may be licensed under this regulation, or under Appendix A of 10 CFR Part 100, and use the approach described in PWROG-16043 as part of their demonstration that the criteria are met. Thus, the NRC staff considered whether the PWORG-16043 approach would be inconsistent with these criteria. The text has been revised to include Appendix A of 10 CFR Part 100 (which contains similar requirements) and to clarify that the specific regulatory requirements are site-specific.
2	16	45	Typo – "forr" should be "for".	The staff agrees, and the proposed change was incorporated as-is.
3	17	22-28	The proposed changes on Page 16, Lines 13 – 19 of the PWROG comments on the original DSE were not incorporated. The NRC response to PWROG Comment 1 was not incorporated (refer to NRC response matrix on PWROG comments.)	The staff agrees. This was an oversight, and the prior recommendations were incorporated as-is.

Attachment

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SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-16043-P, REVISION 2, "PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09: 'IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH' FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS" (EPID: L-2018-TOP-0021) DATED OCTOBER 31, 2019

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ADAMS Accession Nos.:
ML19302H038 -Package
ML19242B695 -Cover Letter
ML19242B646 Disposition Table
ML19302F448 -Final SE Enclosure *concurrence via email **NRR-106**

OFFICE	NRR/DORL/LLPB/PM	NRR/DORL/LLPB/LA	NRR/DSS/SNPB*	NRR/DORL/LLPB
NAME	JDrake*	DHarrison	RLukes	DMorey
DATE	10/29/2019	10/30/2019	10/30/2019	10/31/2019

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR TOPICAL REPORT PWROG-16043-P, REVISION 2,

"PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09:

'IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH'

FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS"

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

1.0 INTRODUCTION

By letter dated February 1, 2017 (Reference 1), the Pressurized Water Reactor (PWR) Owners Group (PWROG or the applicant), submitted to the U.S Nuclear Regulatory Commission (NRC) staff for review licensing topical report (TR) PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09. 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" (Reference 2, henceforth referred to as the TR). Subsequent letters dated March 27, 2018, May 15, 2018, and May 15, 2018 (References 3, 4, and 5, respectively), provided additional information that supplemented the information provided in Reference 2. The TR will be used as the basis for determining fuel assembly characteristics and damping coefficients at End of Life (EOL) conditions for input into plant seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods described in WCAP-9401-P-A (Reference 6) and CENPD-178(P), Rev. 1-P (Reference 7), to assess the structural integrity of fuel assemblies under faulted condition loads.

2.0 BACKGROUND

Seismic and LOCA events can result in external forces applied to the fuel assemblies (e.g., shaking and/or vibratory forces) Therefore, applicants must evaluate the fuel assembly structural response under these conditions to ensure that regulatory requirements are met with respect to control rod insertability and core coolability. In particular, the spacer grid performance is assessed to determine if plastic deformation is expected to occur, and the fuel assembly vibration behavior is quantified. Most PWR plants currently utilize the NRC approved testing and analysis methodologies described in References 6 and 7 for Westinghouse and CE fuel designs, respectively

The NRC reviewed and approved References 6 and 7 based on the regulatory guidance provided in Appendix A to Section 4.2 of the Standard Review Plan (SRP or Reference 8) One assumption in the SRP Section 4 2 Appendix A guidance at the time, which is also in the current revision from 2007, is that beginning of life (BOL) is the time at which the crushing load for the spacer grids would be expected to be at a minimum This assumption was based on the fact that irradiation tends to cause strengthening in metals and alloys in addition to embrittlement Other effects that arise due to use in a reactor may include growth, cladding creep, and corrosion The increase in strength was expected to more than offset the other effects associated with irradiated grids Since applicants typically verify that the maximum load experienced by the spacer grids during LOCA and seismic events will not exceed the crushing load, use of BOL characteristics was considered to be conservative.

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Operating experience that came to light in the mid-2000s led the NRC staff to question the assumption that the spacer grid structural performance during LOCA and seismic events would not degrade significantly as a result of irradiation. The NRC subsequently issued Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" (Reference 9) This IN lists several factors that can affect the structural strength of the spacer grids and singles out spacer grid spring relaxation as one that can have a significant effect on the fuel assembly mechanical characteristics and the spacer grid strength. While no specific action or response was required as a result of the IN, the NRC indicated that recipients would be expected to review the information for applicability and consider appropriate action to avoid similar problems.

This TR is the applicant's proposed approach to generically address the issue identified in the IN for licensees that use Westinghouse or CE fuel. This TR will be used as the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant specific seismic and LOC analyses that will be performed in accordance with the current NRC approved methods described in References 6 and 7, to assess the structural response of fuel assemblies under faulted condition loads. Crediting flowing water damping ratios in a similar manner to the NRC approved still water damping ratios (as described in References 6 and 7) provides a means for licensees to recover margin lost due to the effect of spacer grid spring relaxation on the fuel assembly mechanical characteristics.

In summary, the existing NRC approved testing and analysis methodologies will continue to be used, with all previously established limitations and conditions, however, this TR provides the basis for determining fuel assembly characteristics and damping coefficients to address potential fuel assembly structural performance issues as a result of irradiation.

3.0 REGULATORY EVALUATION

Title 10, "Energy," of the *U.S. Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," contains requirements for the emergency core cooling system (ECCS) at commercial power plants In particular, 10 CFR 50.46(b)(4) requires that "[c]alculated changes in core geometry shall be such that the core remains amenable to cooling." Any failure in the structural integrity of the fuel assemblies will typically change the core geometry, and the possibility needs to be evaluated.

The regulation at 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC)-10, "Reactor design," states that "[t]he reactor core shall be designed with appropriate margin to assure that specified fuel design limits are not exceeded during...anticipated operational occurrences " Within the context of LOCA and seismic events, this is implicitly addressed by ensuring adequate core coolability.

The regulation at 10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," states that "[t]he reactivity control systems shall be designed to... reliably [control] reactivity changes. " One of the primary reactivity control systems at current WEC and CE PWR plants is the rapid insertion of control rods to add sufficient negative reactivity to shut down the reactor. Reliable operation of this reactivity control system is conditional on the capability to insert the control rods. Vibrations or structural deformations may impede the control rod movement and need to be evaluated

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The regulation at 10 CFR Part 50, Appendix A, GDC 35, "Emergency core cooling," restates the requirement to maintain adequate emergency core cooling capability, which can be affected by the core geometry as discussed in 10 CFR 50 46(b)(4) (see above)

The regulation at 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," requires safety-related structures, systems, and components (SSCs), including reactor fuel, to be designed to withstand natural phenomena (such as earthquakes) without a loss of capability to perform safety functions. This GDC also requires consideration of "appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena." For example, a LOCA may be caused by a seismic event, so consideration of the effects from a combination of these two events may be appropriate.

Appendix S of 10 CFR Part 50 and Appendix A of 10 CFR Part 100 provide additional guidance for seismic events, and defines the Safe Shutdown Earthquake (SSE), Operating Basis Earthquake (OBE), and safety requirements for relevant SSCs In general, criteria should be defined for each SSC to ensure its functional capabilities during each event indicated by the regulatory requirements (typically OBE, LOCA+SSE, and SSE-only, though other combinations may be considered). These requirements are not explicitly addressed by the methodologies submitted for NRC review, however, the overall methodology that PWROG-16043 will supplement may be used by licensees to demonstrate that these requirements are met (if applicable). Therefore, the NRC staff considered the potential impact of PWROG-16043 on how these requirements would be met

In summary, the NRC staff used the applicable acceptance criteria defined in Section 4.2, Appendix A of the SRP , otherwise known as NUREG-0800 (Reference 8), in its review of the TR. Since the TR provides an alternate approach to produce parameters for use with existing methodologies, the scope of the NRC staff review was limited to the testing protocols and analysis approaches described in the TR to develop the aforementioned parameters, and to verify the applicability of the existing methodologies when using the parameters developed with the new approaches. The primary criteria are related to ensuring that core coolability and control rod insertability are maintained

4.0 TECHNICAL EVALUATION

The intent of the TR is to develop the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with current NRC approved methods in References 6 and 7 by focusing solely on the specific parameters that would be Impacted by the EOL issues identified in IN 2012-09 (Reference 9). As such, the TR narrowly focuses on three primary parameters·

1. The allowable grid impact strength [
   ]
2. The fuel assembly modal frequencies [

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] and

3. The fuel assembly flowing water damping ratio, [
]

Section 15.0.2 of the SRP, "Review of Transient and Accident Analysis Methods," (Reference 10) provides guidance for review of transient and accident analysis methods This guidance is not directly applicable to this TR, since the analysis methods described in References 6 and 7 are not being modified, only the empirical determination of key input parameters Therefore, the NRC staff review of the TR only focused on the two specific areas described in SRP Section 15.0 2 that are relevant to the applicability of the analysis methods when a different approach is used to develop input parameters, as described below.

1 Evaluation methodology – the proposed testing and data analysis approach, including any potential limitations to their applicability

2 Uncertainty analysis – the applicant's evaluation and propagation of uncertainties in the analysis of test data to obtain recommended values for the key parameters.

In addition, the NRC staff considered whether the applicant provided adequate quality assurance (QA) and documentation support for the proposed approach for addressing the EOL effects on spacer grids This aspect is not explicitly discussed in detail for this safety evaluation (SE) because the documentation of the proposed approach is captured by the documents reviewed by the NRC during an audit dated October 17, 2017 (Reference 11) and that were found to have been appropriately summarized or otherwise characterized in the TR. The testing was performed under the auspices of the same QA program for testing previously performed to determine the key parameters for BOL grids and still water damping, which is acceptable. As such, the NRC staff acceptance of the adequacy of the applicant's test protocol and uncertainty analyses implicitly includes acceptance of the applicant documentation associated with that area.

4.1 EOL Grid Simulation

This TR discusses the test protocol used for the characterization of the impact of irradiation on the spacer grids SRP Section 4.2 Appendix A (Reference 8) cites several possible irradiation-related effects relevant to spacer grids and concludes that the combined impact would not be expected to lead to a more conservative result. This logic rests mainly on the fact that the significant increase in yield strength for the spacer grid material will more than offset the relatively minor effects from the remaining effects. As described in IN 2012-09 (Reference 9), operating experience has shown that spacer grid spring relaxation can have a significant adverse effect on spacer grid strength and fuel assembly mechanical characteristics. [

] Other than grid spring relaxation, the basic assessment in SRP Section 4 2 Appendix A that irradiation-related effects are bounded by the increase in the yield strength of the spacer grid material continues to be applicable [

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]

[

] As discussed in the previous paragraph, the NRC staff found that the focus on the grid spring relaxation phenomenon as the key driver for the non-conservative behavior identified in spacer grids at EOL relative to BOL is appropriate. However, the material and geometry impact of the thermal relaxation process must be reasonably similar to the irradiation-induced impacts that are being simulated

[

] Therefore, the NRC staff requested additional information from the applicant regarding the thermal relaxation procedure used to produce the simulated EOL grids. [

] The applicant's response also confirmed that the material structural characteristics of the simulated EOL grids are the same, or slightly conservative, relative to the BOL grids.

[

] There are some situations where a spacer grid is exposed to a strongly non-uniform neutron flux, such as fuel assembly loading locations at or near the core periphery. The NRC staff asked the applicant to address the potential impact on the grid failure mechanism due to non-random gradients in gap size that may be correlated with steep neutron flux gradients. [

]

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Finally, Section 2 1 of the TR described how the target average gap size was determined for a given spacer grid [

]

Inadequate information was given in the TR to define the area of applicability for extrapolation of a given set of PIE data to the general population of EOL grid spacers of the same design, so the NRC staff requested that the applicant characterize how PIE data sets are generally defined in order to achieve their intended purpose.

The applicant responded in Reference 4 with an explanation of the statistical technique underlying their determination of a target gap size for the simulated EOL grids [

] this is a reasonably conservative approach to ensure that the average gap sizes for the simulated EOL grids will bound the average gap sizes for irradiated grids.

[

] The NRC staff agrees; however, the applicant did not describe how the rod burnups associated with the PIE measurements would be used to define the area of applicability for fuel assemblies qualified using this approach. In a separate RAI response (RAI-2, documented in Reference 4), the applicant provided information that shows that the variation in gap sizes for varying burnups near EOL can be expected to be minor relative to the inherent randomness in gap sizes within a grid. In addition, the NRC staff noted that the protocol described in Reference 7 for testing of CE design fuel assemblies includes modeling for both BOL and EOL grids [

] Consistent with this assessment, the results from the testing discussed in Sections 4.2 and 4.3 of this SE show [

]

Therefore, any variations in burnup for the fuel assemblies used to obtain PIE measurements relative to the overall population of fuel assemblies being qualified using this approach would not result in a significant difference in average gap size, certainly, much less than the inherent conservatism in the margin between the average measured gap sizes and the target gap size for the simulated EOL grids

The NRC staff found that the subject TR described an acceptable approach to produce simulated EOL grids for testing that accounts for the range of expected variation from irradiated

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EOL grids. [

] As a result, the NRC staff found the proposed approach to generate simulated EOL grids for use in testing in lieu of irradiated grids to be acceptable

4.2 Spacer Grid Impact Strength

Sections 2.2 and 2.3 of the subject TR discuss the application of the approved testing and data analysis protocol from References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. In all respects, the testing and data analysis applications were consistent with References 6 and 7, [

] The NRC staff understanding of the approval request from the applicant is that this additional criterion was provided merely for demonstration purposes and was not submitted as a change to how the grid impact strength is determined in Reference 6. In response to a RAI from the NRC staff (Reference 3), the applicant confirmed that this was the case Therefore, this application was judged to be acceptable solely for the purpose of providing a more consistent basis for comparing P(crit) for Westinghouse and CE fuel designs

The simulated EOL grids contain [

] The NRC staff verified by inspection of the applicant's test documentation that the failure mechanism for the simulated EOL grids was the same as that for the BOL grids Therefore, [

] As discussed in Section 4 2 of this SE, [

]

The NRC staff verified that the previously approved testing and data analysis protocols in References 6 and 7 were appropriately applied to the simulated EOL grids In addition, the NRC staff found reasonable assurance exists that the aforementioned test protocols remain applicable to the geometry of the simulated EOL grids Therefore, the NRC staff found the approach for determining P(crit) to be acceptable for use in analysis of the simulated EOL grids.

4.3 Fuel Assembly Mechanical Characteristics

Section 3 of the TR discusses the application of the approved testing and data analysis protocols in References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. The TR states that "[t]he same test protocol has been previously applied to current Westinghouse and CE PWR fuel designs for BOL conditions," and that "[t]he test protocols are described in NRC-approved TRs..." with a citation to References 6 and 7. Therefore, the TR clearly characterizes the testing procedure for the simulated EOL grids to be identical to the previously approved testing procedure described in References 6 and 7, with the exception that the grids are simulated EOL grids as discussed in Section 4.1 of this SE

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The testing protocols described in References 6 and 7 are primarily tests conducted on the structural members of the fuel assembly and the spacer grids, with no tests directly impacting the fuel rods At BOL, the grid springs exert a frictional force on the fuel rods, so the spacer grids and fuel rods are mechanically coupled to some extent. During the fuel assembly vibration tests, the fuel rods contribute to the fuel assembly mechanical performance by virtue of this mechanical coupling. [

]

[

]

4.4 Procedure to Determine Flowing Water Damping Ratios

Section 4 of the TR describes the test protocol for determining the fuel assembly flowing water damping ratios and apply them in lieu of previously approved still water damping ratios to characterize the fuel assembly mechanical behavior during seismic and LOCA events. Since the damping ratio due to flowing water is expected to be higher than that for still water, this approach could help recapture margin lost due to the impact of grid spacer relaxation on the fuel assembly stiffness. [

]

Sections 4.1 through 4.3 describe the test apparatus and data collection performed to support an empirical determination of the flowing water damping ratios. [

] However, the hydraulic characteristics for the fuel assembly are well characterized based on prior testing. [

] Since the loss coefficients for the fuel assembly designs have been approved by the NRC for use in other analyses and would not be expected to vary significantly as a result of the use of simulated EOL grids, this approach for determining flow velocities through the fuel assembly is acceptable.

The existing test protocols, most notably the Reference 7 protocol for CE fuel, [

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] Testing performed on similar fuel assembly designs using a range of different approaches, as documented in References 14 and 15, yield consistent results. [

] This shows that the proposed approach discussed in this TR yields results consistent with what was approved in References 13 and 15 (Reference 14 was incorporated in the approved TR represented by Reference 13).

The flowing water damping ratio correlation was developed based [

] Therefore, there will be no inconsistency in the application of damping ratios for fuel assemblies at different burnup conditions.

Based on the data collected from the tests, a damping ratio was determined for each test based on classical vibration theory [

] Section 4 5 of the TR presents results from the tests. One of the most important conclusions that can be observed directly from the test results is that [

] Since the use of lower damping ratios in developing the correlation is conservative, this was an acceptable choice to make

Section 4 6 of the TR discusses the data analysis approach used to determine bounding correlations for each fuel assembly design. This approach can be summarized thus: [

] The overall approach appears to capture the relevant dependencies, however, there is no propagation of the uncertainties due to scatter in data through the steps noted above [

]

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The applicant responded in Reference 5 with information indicating that the fitting approach used to determine the bounding curve was fundamentally a best estimate approach to derive the 600 °F curve based on the selected data set [ ]

[ ]

Finally, Section 4.7 proposes use of a flowing water damping ratio correlation based on the [ ] fuel assembly design as a generically bounding correlation that may be used with any fuel assembly design without further justification. The procedure discussed above may be used to develop fuel assembly design specific correlations, but the [ ] correlation is proposed for use as a bounding curve for all Westinghouse and CE fuel designs The justification provided is that the [ ] fuel assembly design proposed for the [ ] reference plant contains a number of significant design differences, but test results show that the flowing water damping ratio is very similar to the [ ] fuel. The CE fuel design tested had [ ] This behavior is bounded by the [ ] correlation, so this is acceptable. However, [ ] Therefore, the similarity in results is not surprising

In order to establish that the proposed correlation can be used as a generic bounding curve, its applicability must be limited to spacer grids with very similar geometry characteristics This is accomplished via a condition to the TR. Information submitted in References 14 and 15 provide information for other PWR fuel assembly designs that suggests that, in fact, the [ ]

As long as the geometry characteristics of the spacer grids associated with a different fuel assembly do not differ significantly from the [ ], the NRC staff finds that reasonable assurance exists that other fuel assembly designs will have flowing water damping ratios near or above the proposed bounding curve. The proposed application includes use of a

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minimum value for the analysis duration rather than a more realistic average value, which incorporates some additional conservatism that offsets the potential for slightly lower flowing water damping ratios for some fuel assembly designs relative to the proposed bounding curve.

Based on the information provided in the TR, as supplemented by responses to requests for additional information from the NRC staff, the testing protocol and data analysis described to determine appropriate flowing water damping ratios were determined to be appropriate for their intended purpose. In addition, [

] This latter condition was captured in Section 5.0.

4.5 Analytical Application of the Flowing Water Damping Ratios

Sections 4 8 and 4.9 of the TR describe when and how the flowing water damping ratios can be utilized in seismic and LOCA analyses, respectively. The primary parameter used to establish the appropriate value for the flowing water damping ratio is the fluid velocity through the fuel assembly. For a given plant, this parameter is directly correlated with the core flow. Therefore, the discussion in the TR primarily focuses on the characterization of a bounding core flow for any given time of interest during the event being analyzed. Once an appropriate value is determined, then plant-specific information can be used to establish an appropriate flow velocity to use with the flowing water damping ratio correlation. [

] In general, since lower flow velocities result in lower flowing water damping ratios, any factor that may lead to a reduction in the core flow rate will provide more conservative results. For a given analysis, [

]

For the seismic analysis, two key assumptions are made to minimize the total core flow. First, [

]

Secondly, [

] At this time, the flowing water damping ratio will be at a minimum, and

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lower than the average flowing water damping ratio for the interval Since these assumptions both act to minimize the flowing water damping ratio, they are conservative

For the LOCA analysis, the core flow rates are to be obtained directly from the LOCA analyses, as long as axial flow is maintained. [

] As a result, the NRC staff finds that the LOCA analysis conditions are an acceptable source for a bounding core flow rate for the purpose of determining flowing water damping ratios.

A second limitation of the flowing water damping ratios is that the data used as a basis for the correlation were based on single phase liquid flow through a fuel assembly. The conditions under which the flowing water damping ratios are expected to be credited—seismic events and the first ~1 second of a LOCA event—are not expected to involve two phase flow in the core. However, the TR does not explicitly limit the use of flowing water damping ratios to single phase flow conditions, so a limitation was included in Section 5 0 to ensure that, if credit for flowing water damping is applied to conditions that deviate from expectations, the correlation will not be used outside the bounds of its applicability.

The NRC approval of Reference 13 included review of information demonstrating that the Westinghouse models were capable of capturing the dynamic behavior of fuel assemblies for pluck response inside a flow loop, for the vibration range of interest Since the flowing water damping ratios are very similar for the RFA/RFA-2 curves being proposed for use as a bounding curve for all fuel assembly designs and the Reference 13 fuel design contained a similar spacer grid design, this finding is applicable to the subject LTR as well. However, without further validation, the dynamic models cannot be assumed to maintain reasonable accuracy for damping ratios that go significantly beyond the current area of applicability. Therefore, any use of damping ratios significantly higher than the proposed bounding curve must be supported by a demonstration that the analytical models remain valid for the higher damping regime A limitation was included in Section 5 0 to ensure that this potential limitation of the analytical models is addressed, if necessary

The guidance provided in the TR to credit flowing water damping in seismic and LOCA analysis was reviewed by the NRC staff and determined to produce acceptably conservative results for the expected analysis conditions. Therefore, the NRC staff finds the proposed application of flowing water damping credit for evaluation of fuel assembly mechanical behavior during seismic and LOCA events to be acceptable.

4.6 Legacy Issues

There are a number of potential issues with the previously approved methodologies described in References 6 and 7. These issues did not exist at the time that the methodologies in References 6 and 7 were approved, however, more recent fuel assembly designs have incorporated features such as thinner spacer grid straps that may undergo more plastic deformation prior to failure. The NRC staff has not retroactively required licensees to address them based on the inherent conservatism within the previously approved methodologies and typical margins for licensing seismic analyses. However, the potential use of flowing water damping credit represents a more realistic (i.e., less conservative) approach. As such, licensees should address the following issues before they reduce conservatism in their licensing

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basis analyses that utilize the methodologies from References 6 and 7. In some cases, these issues have already been addressed for existing fuel assembly designs

[

]

In order to ensure that the overall analysis remains conservative, a limitation was included in Section 5 0 to restrict use of flowing water damping credit unless information is provided to address the above issues. The limitation can be resolved by providing information to demonstrate that any predicted loads on the guide tubes and spacer grid cage remain within the elastic region This ensures that (1) through (3) are directly satisfied, and implicitly ensures that (4) is met by ensuring that safety related components are capable of performing their safety function under the combined effects of SSE and the accident loads for which their function is required.

As discussed above, the NRC staff identified some technical issues that are not explicitly addressed by the currently approved methodology They may have been addressed for current fuel assembly designs, however, the use of a more realistic flowing water damping ratio represents a reduction in conservatism for the damping ratio approach relative to the previously approved approach Therefore, the NRC staff is imposing limitations and conditions to ensure that the overall conservatism of the analysis is acceptable.

5.0 LIMITATIONS AND CONDITIONS

Some limitations and conditions are necessary to ensure that the approaches discussed in the TR is limited to the applications for which it is valid, and to ensure that the overall analysis methodology remains conservative. These limitations and conditions are listed below

1. [

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]

6.0 CONCLUSIONS

The applicant submitted a TR that will be used as the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods as described in References 6 and 7, to assess the structural integrity of fuel assemblies under faulted condition loads. The following conclusions are provided here in summary as they apply to licensees who may want to adopt the TR to address the effect of irradiation on the mechanical properties of fuel assemblies.

Since the TR is not proposing any change to the previously approved testing and analysis methods for seismic and LOCA events, the NRC staff performed a graded review of the TR that took into consideration the fact that most aspects of the methods that this TR is intended for use with have already been addressed as part of prior NRC reviews The applicant requested approval of seven specific items identified in Section 1.4 of the TR

The NRC staff examined the proposed approach to produce simulated EOL spacer grids and determined that the simulated EOL spacer grids would adequately capture the non-conservative impacts due to irradiation. The staff also determined that the [

] The NRC staff's findings were based primarily on the specific material type (zirconium alloy) and general grid design covered by the information presented in the TR, [

]

The use of flowing water damping ratios is not an entirely new approach to develop more realistic parameters that help mitigate the impact of vibratory loads, because it is similar to what was approved by the NRC for the AP1000 (Reference 13). However, this is the first time that it is being applied more generically to Westinghouse and CE fuel In particular, the applicant is proposing the use of a bounding curve that is applicable to all spacer grids used in Westinghouse and CE fuel, along with a general approach that can be used to generate fuel design specific curves The staff reviewed the information submitted in the TR along with responses to requests for additional information, and determined that the approach was

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appropriate for both purposes. Additionally, the guidance provided for utilization of flowing water damping ratios in seismic and LOCA analyses was found to be appropriate for their intended use, with the limitations that. (1) the flowing water damping ratios are only valid for single phase liquid flow, and (2) the dynamic models used to predict the fuel assembly response under vibratory and damping loads must be verified to remain reasonably accurate for higher damping regimes (Limitations and Conditions 2 and 4)

The NRC staff also acknowledged some legacy issues that have not previously been addressed by the NRC staff due to their low risk significance, based on the inherent conservatisms within the analysis methods described in References 6 and 7. Consequently, the NRC staff finds that any reduction in analytical conservatism should not be made without addressing these legacy issues, as discussed in Section 4.6. The use of flowing water damping ratios represents one such reduction in analytical conservatism, therefore, a condition for use of the new damping ratios is that the legacy issues need to be addressed (Limitation and Condition 3).

In summary, the NRC staff finds that the information provided in the TR and responses to NRC staff RAIs adequately demonstrates that the proposed approach to address EOL effects on spacer grids and to recover margin through credit for flowing water damping are acceptable for use with existing methods that the NRC has previously found to be acceptable for analysis of fuel assembly structural behavior during seismic and LOCA events The NRC staff approval of this TR extends to all Westinghouse and CE fuel designs, contingent on adherence to the limitations and conditions set forth in Section 5.0.

7.0 <u>REFERENCES</u>

1 PWROG letter OG-17-12, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Submittal of PWROG-16043-P, Revision 2, 'PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169R2," February 1, 2017 (ADAMS Accession No. ML17039B050)

2 PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 2017 (ADAMS Package Accession No. ML17039B061)

3 PWROG letter OG-18-62, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request for Additional Information, RAIs 4 and 5 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09· 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," March 27, 2018 (ADAMS Accession No. ML18100A053)

4. PWROG letter OG-18-104, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request for Additional Information, RAIs 1, 2, and 3 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09. 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," May 15, 2018 (ADAMS Accession No ML18143B462)

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5. PWROG letter OG-18-105, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request for Additional Information, RAI 6 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," May 15, 2018 (ADAMS Accession No. ML18144A760)

6. WCAP-9401-P-A, Revision 0, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," September 1981 (ADAMS Accession No. ML090280466 (Non-Publicly Available))

7. CENPD-178(P), Revision 1-P, "Structural Analysis of Fuel Assemblies for Seismic & LOCA Loading," August 1981 (ADAMS Accession No ML14122A086 (Non-Publicly Available))

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 4.2, Revision 3, "Fuel System Design," March 2007 (ADAMS Accession No. ML070740002)

9. NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," dated June 28, 2012 (ADAMS Accession No. ML113470490)

10. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 15.0.2, Revision 0, "Review of Transient and Accident Analysis Methods," March 2007 (ADAMS Accession No. ML070820123)

11. NRC letter from Brian Benney, Senior Project Manager, Licensing Processes Branch, Division of Policy and Rulemaking, USNRC, to Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, re. "Summary Report for the October 17, 2017, Audit in Support of the Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 8, 2018 (ADAMS Accession No. ML17326A003)

12. Framatome ANP, Inc. letter NRC-03 051, James F Mallay, Director, Regulatory Affairs, Framatome ANP, Inc., to USNRC document control desk, re. "Closure of Interim Report 02-002, 'Spacer Grid Crush Strength – Effects of Irradiation,'" August 8, 2003 (ADAMS Accession No ML032240425)

13. WCAP-17524-P/NP-A, Revision 1, "AP1000 Core Reference Report," May 2015 (ADAMS Accession No. ML15180A175)

14. Westinghouse letter LTR-NRC-13-26, James A. Greshman, Manager, Regulatory Compliance, Westinghouse Electric Company, to USNRC document control desk, re: "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor (Proprietary/Non-Proprietary)," April 30, 2013 (ADAMS Accession No. ML13128A017)

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15 Framatome Inc report ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018 (ADAMS Package Accession No. ML18144A816)

Principal Contributor: Scott Krepel, NRR/DSS/SNPB

Date. May 17, 2019

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PWR Owners Group
United States Member Participation* for PA-ASC-1169, Revision 4

Utility Member	Plant Site(s)	Participant Yes	Participant No
Ameren Missouri	Callaway (W)	X	
American Electric Power	D C. Cook 1 & 2 (W)	X	
Arizona Public Service	Palo Verde Unit 1, 2, & 3 (CE)	X	
Dominion Connecticut	Millstone 2 (CE)		X
	Millstone 3 (W)	X	
Dominion VA	North Anna 1 & 2 (W)	X	
	Surry 1 & 2 (W)	X	
Duke Energy Carolinas	Catawba 1 & 2 (W)		X
	McGuire 1 & 2 (W)		X
	Oconee 1, 2, & 3 (B&W)		X
Duke Energy Progress	Robinson 2 (W)		X
	Shearon Harris (W)		X
Entergy Palisades	Palisades (CE)		X
Entergy Nuclear Northeast	Indian Point 2 & 3 (W)	X	
Entergy Operations South	Arkansas 1 (B&W)		X
	Arkansas 2 (CE)	X	
	Waterford 3 (CE)	X	
Exelon Generation Co. LLC	Braidwood 1 & 2 (W)	X	
	Byron 1 & 2 (W)	X	
	TMI 1 (B&W)		X
	Calvert Cliffs 1 & 2 (CE)		X
	Ginna (W)		X
FirstEnergy Nuclear Operating Co.	Beaver Valley 1 & 2 (W)		X
	Davis-Besse (B&W)		X
Florida Power & Light \ NextEra	St. Lucie 1 & 2 (CE)		X
	Turkey Point 3 & 4 (W)		X
	Seabrook (W)		X
	Pt Beach 1 & 2 (W)		X
Luminant Power	Comanche Peak 1 & 2 (W)	X	

Omaha Public Power District	Fort Calhoun (CE)		X
Pacific Gas & Electric	Diablo Canyon 1 & 2 (W)	X	
PSEG – Nuclear	Salem 1 & 2 (W)		X
South Carolina Electric & Gas	V.C Summer (W)	X	
So. Texas Project Nuclear Operating Co.	South Texas Project 1 & 2 (W)		X
Southern Nuclear Operating Co.	Farley 1 & 2 (W)	X	
	Vogtle 1 & 2 (W)	X	
Tennessee Valley Authority	Sequoyah 1 & 2 (W)		X
	Watts Bar 1 & 2 (W)	X	
Wolf Creek Nuclear Operating Co.	Wolf Creek (W)	X	
Xcel Energy	Prairie Island 1 & 2 (W)		X

Note*: Project participants as of the date the final deliverable was completed On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.

PWR Owners Group
International Member Participation* for PA-ASC-1169, Revision 4

Utility Member	Plant Site(s)	Participant Yes	Participant No
Asociación Nuclear Ascó-Vandellòs	Asco 1 & 2 (W)		X
	Vandellos 2 (W)		X
Axpo AG	Beznau 1 & 2 (W)		X
Centrales Nucleares Almaraz-Trillo	Almaraz 1 & 2 (W)	X	
EDF Energy	Sizewell B (W)		X
Electrabel	Doel 1, 2 & 4 (W)		X
	Tihange 1 & 3 (W)		X
Electricite de France	58 Units	X	
Eletronuclear-Eletrobras	Angra 1 (W)		X
Eskom	Koeberg 1 & 2 (W)		X
Hokkaido	Tomari 1, 2 & 3 (MHI)		X
Japan Atomic Power Company	Tsuruga 2 (MHI)		X
Kansai Electric Co., LTD	Mihama 1, 2 & 3 (W)		X
	Ohi 1, 2, 3 & 4 (W & MHI)		X
	Takahama 1, 2, 3 & 4 (W & MHI)		X
Korea Hydro & Nuclear Power Corp.	Kori 1, 2, 3 & 4 (W)		X
	Hanbit 1 & 2 (W)		X
	Hanbit 3, 4, 5 & 6 (CE)		X
	Hanul 3, 4 , 5 & 6 (CE)		X
Kyushu	Genkai 1, 2, 3 & 4 (MHI)		X
	Sendai 1 & 2 (MHI)		X
Nuklearna Electrarna KRSKO	Krsko (W)		X
Ringhals AB	Ringhals 2, 3 & 4 (W)		X
Shikoku	Ikata 1, 2 & 3 (MHI)		X
Taiwan Power Co.	Maanshan 1 & 2 (W)		X

Note*. Project participants as of the date the final deliverable was completed On occasion, additional members will join a project. Please contact the PWR Owners Group Program Management Office to verify participation before sending this document to participants not listed above.

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United States (U.S.) Nuclear Regulatory Commission (NRC) Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," was issued in June 2012. The IN discusses that based on recent operating experience, the crush strength of the fuel assembly spacer grids may decrease during the life of a fuel assembly. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (SRP) Section 4.2, "Fuel System Design," Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," infers that fuel spacer grid strength only needs to be considered at Beginning-Of-Life (BOL) conditions with respect to evaluating fuel structural integrity. The Westinghouse methodologies for assessing the structural integrity of fuel assemblies under faulted condition loads (seismic and LOCA) are contained in two NRC-approved topical reports (TRs), WCAP-9401-P-A, "Verification Testing and Analyses of the 17X17 Optimized Fuel Assembly," and CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading." The plant-specific analyses are currently performed with fuel assembly spacer grid characteristics at BOL conditions based on SRP Section 4.2, Appendix A.

This TR addresses the issue identified in NRC IN-2012-09 by applying the approach that was used to address the EOL effects for the **AP1000®**[1] Core Reference Report APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report". The tests utilized the NRC-approved methodologies contained in WCAP-9401-P-A and CENPD-178-P, Rev.1-P with simulated EOL grids. Additionally, flowing water damping testing was performed and flowing water damping within the NRC-approved methodologies can be credited consistent with the approach used to address EOL effects in the AP1000 Core Reference Report that was approved by the NRC. Testing was performed on two fuel designs, the Westinghouse 17x17 Robust Fuel Assembly-2 and the Combustion Engineering (CE) CE16NGF™[1], and is discussed in this TR.

This TR discusses the applicability for determining fuel assembly characteristics and damping coefficients at EOL conditions and the aspects for which NRC approval is requested. This TR does not revise and or modify the current grid and fuel assembly test methods, or the fuel assembly seismic and LOCA analysis methodologies, processes and codes that were previously approved by NRC.

[1] **AP1000 and CE16NGF** are 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. Other names may be trademarks of their respective owners.

1 INTRODUCTION AND ASPECTS REQUESTED FOR APPROVAL

1.1 INTRODUCTION

United States (U.S.) Nuclear Regulatory Commission (NRC) Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," was issued in June 2012 (Reference 1-1). The IN discusses that based on recent operating experience, the crush strength of the fuel assembly spacer grids may decrease during the life of a fuel assembly. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (SRP) Section 4.2, "Fuel System Design," Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" (Reference 1-2), infers that fuel spacer grid strength only needs to be considered at Beginning-Of-Life (BOL) conditions with respect to evaluating fuel structural integrity. The Westinghouse methodologies for evaluating the structural integrity of fuel assemblies under faulted condition loads (seismic and LOCA) are contained in two NRC-approved topical reports (TRs), WCAP-9401-P-A, "Verification Testing and Analyses of the 17X17 Optimized Fuel Assembly," and CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading" (References 1-3 and 1-4, respectively). The plant-specific analyses are currently performed with fuel assembly spacer grid characteristics at BOL conditions based on SRP Section 4.2, Appendix A.

For a fuel assembly with zirconium alloy grids, the irradiation effects due to rod diameter creep, grid spring relaxation and grid growth, also called the End-of-Life (EOL) conditions, can reduce grid spring preload, and allow small gaps between the rod and grid supports to form. The irradiation effects can reduce the zirconium alloy grid impact strength, and also, reduce the fuel assembly bundle stiffness and natural frequencies The irradiation effects could potentially increase the grid impact loads and fuel assembly component stresses during seismic and LOCA events To address this issue, fuel assembly damping in flowing water can be used to offset the EOL irradiation effects.

To address NRC IN 2012-09 for the Westinghouse and Combustion Engineering (CE) PWR fuel designs, the PWROG Analysis Committee and Westinghouse proactively initiated a program, the result of which are documented in this topical report. This topical report is based on the NRC-approved approach used for the AP1000 Core Reference Report (Reference 1-5) and includes testing of Westinghouse and Combustion Engineering (CE) PWR fuel designs at simulated EOL conditions.

The topical report addresses three key items with respect to evaluating the structural integrity of fuel assemblies under faulted condition loads:

[ ]c

1.2 OVERVIEW OF REPORT CONTENTS

In this TR (PWROG-16043-P), the testing of two fuel designs is discussed. These were the Westinghouse 17x17 Robust Fuel Assembly-2, herein referred to as "RFA/RFA-2," and the CE CE16NGF fuel designs.

Sections 2, 3, and 4 of this TR discuss the three key items listed above.

Section 2 discusses the grid strength tests for two types of grids, the RFA/RFA-2 and CE16NGF. The test setups and methods are based on the NRC-approved test methods discussed in WCAP-9401-P-A (Reference 1-3) and CENPD-178-P, Rev. 1-P (Reference 1-4).

Section 3 discusses the fuel assembly mechanical tests for two fuel assembly designs, the RFA/RFA-2 and CE16NGF. The test setups and methods are based on the NRC-approved test methods discussed in WCAP-9401-P-A (Reference 1-3) and CENPD-178-P, Rev. 1-P (Reference 1-4).

Section 4 discusses the two fuel assembly flowing water damping tests (for the RFA/RFA-2 and CE16NGF) that were performed as part of this PWROG program. The test setup, method, and damping data reduction are consistent with that previously used and approved by the NRC for addressing the EOL effects for the AP1000 plant (Reference 1-5).

These sections discuss the aspects that have been previously approved by the NRC with respect to the testing protocols as described in WCAP-9401-P-A (Reference 1-3) and CENPD-178-P, Rev. 1-P (Reference 1-4). Test protocol, as used in this TR, includes the test setup, testing, and data reduction. These sections also describe the main aspects of the testing that are different from what has been previously approved by the NRC for the current Westinghouse and CE PWR fuel designs. Although these aspects may not have been approved for the current Westinghouse and CE PWR fuel designs, these Sections describe how they have been approved for the AP1000 plant as described in Reference 1-5 and how they have been applied to address IN 2012-09.

1.3 APPLICABILITY OF THIS REPORT

[ ]c

1.4 REQUEST FOR NRC APPROVAL

This submittal does not revise and or modify the current NRC-approved grid and fuel assembly test methods, or the fuel assembly seismic and LOCA analysis methodologies, processes and codes approved by NRC (References 1-3 and 1-4). The purpose of this TR is to only address the issue identified in NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength."

NRC approval of the following is requested:

[ ]c

1.5 REFERENCES

1-1: NRC INFORMATION NOTICE 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," June 28, 2012.
1-2: NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 4.2, "Fuel System Design," Revision 3, March 2007, Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."
1-3: WCAP-9401-P-A, "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," August 1981.
1-4: CENPD-178-P. Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," August 1981.
1-5 APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report," May 2015.

2 ALLOWABLE GRID IMPACT STRENGTH AT EOL CONDITIONS

This section describes the test protocol for determining grid impact strength at EOL conditions. The test protocol uses Westinghouse and CE PWR Fuel design simulated EOL grids for determining grid impact strength at EOL conditions. A more detailed description is provided in the subsequent subsections.

The same test protocol has been previously applied to current Westinghouse and CE PWR fuel designs for BOL conditions

The test protocols are described in NRC-approved TRs WCAP-9401-P-A (Reference 2-1) and CENPD-178-P, Rev. 1-P (Reference 2-2).

The main aspects of the testing described in this section that are different from what has been previously approved by the NRC for current Westinghouse and CE PWR fuel designs are:

[ ] c

Even though these aspects have not been approved for current Westinghouse and CE PWR fuel designs, they have been approved by the NRC for the AP1000 plant as described in TR APP-GW-GLR-153, Rev. 1 (Reference 2-3).

For the testing performed in this program and described in this section, two fuel assembly designs were used: the Westinghouse RFA/RFA-2 and CE16NGF designs.

The results presented in this section are for the purpose of demonstrating the test protocol and for demonstrating the EOL effects to determine the grid strength and grid impact stiffness at EOL conditions. The test protocol is applicable to all Westinghouse and CE PWR fuel designs.

2.1 GRID CELL SIZES AT EOL CONDITION

The grid cell sizes used to simulate the EOL conditions in both the grid impact tests and fuel assembly mechanical tests are mainly based on PIE cell size data. Both the data average and the standard deviation are considered when specifying the target cell size for the test grids. The process of compiling PIE data and specifying target cell size is consistent with that was used for the AP1000 EOL issue that was previously approved by the NRC (Reference 2-3).

[ ] c

a,c

Figure 2-1. Summary of Mid Grid Cell to Rod Gap PIE Data, RFA-2

a,c

2.2 ALLOWABLE GRID IMPACT STRENGTH

The purpose of grid impact testing is to determine the allowable grid impact strength, called crush load P(crit) in SRP (Reference 2-4). SRP states that "the crushing load P(crit) has been suitably selected from the load-versus-deflection curves." The allowable grid impact strength is the maximum grid impact load with small plastic deformation for current Westinghouse and CE grid designs.

[ ]a,c

[ ]a,c

[ ]a,c

2.3 GRID IMPACT TESTS AT EOL CONDITION

The same grid strength test procedures are used for both BOL and EOL conditions. The current Westinghouse test methodologies described in WCAP-9401-P-A (Reference 2-1) and CENPD-178-P, Rev. 1-P (Reference 2-2) are used except for the preparation for test grids as discussed herein.

[ ]a,c

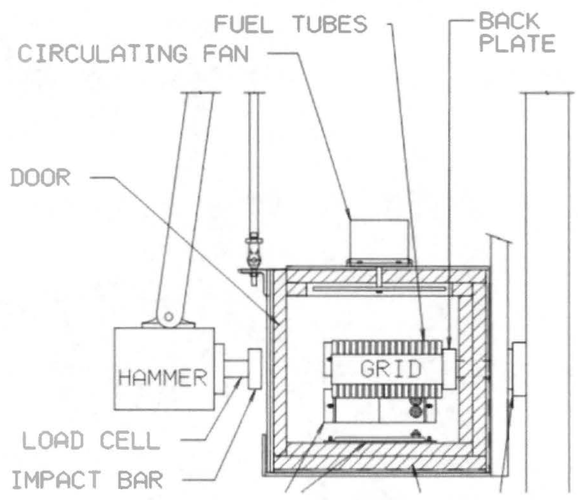


Figure 2-2. Pendulum Grid Impact Test Apparatus

Table 2-1. RFA-2 Mid Grids Test Results for Pendulum Grid Impact Comparisons of at BOL and EOL Conditions

[ ]a,c

For the fuel designs used in CE design cores, the hydraulic long pulse test and drop test are performed. The hydraulic long pulse and the drop test set ups are shown in Figures 2-3 and 2-4, respectively. Both tests are performed at room temperature and test results are scaled to the reactor temperature when they are applied in seismic/LOCA analysis. The test procedures and results application are consistent with CENPD-178-P, Rev. 1-P (Reference 2-2).

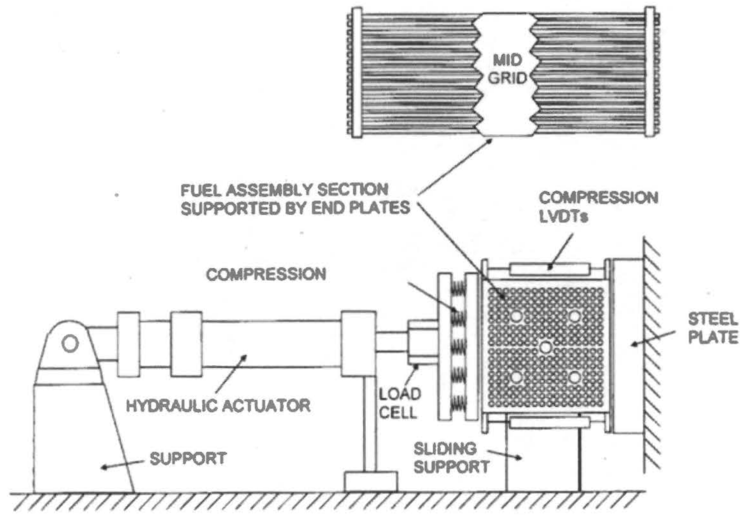


Figure 2-3. Long Hydraulic Grid Impact Test Apparatus

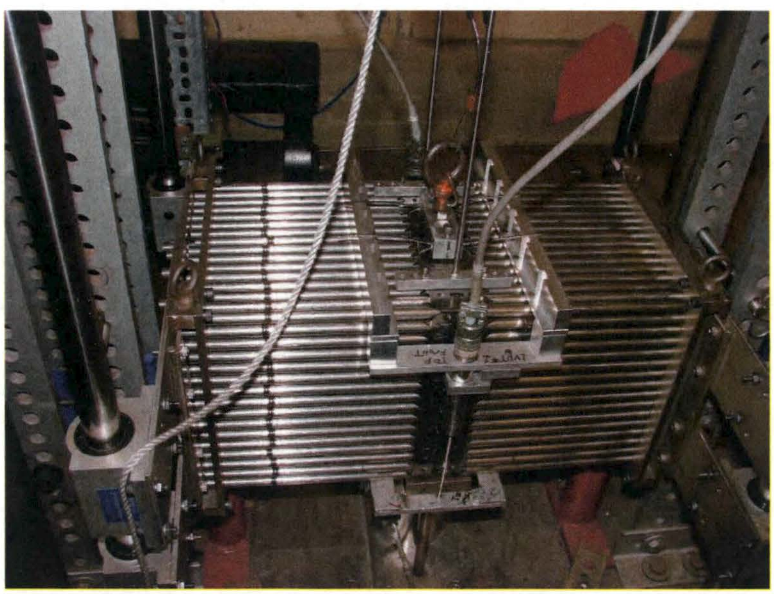


Figure 2-4. One-Sided Impact Grid Strength Apparatus

[ ]a,c

Table 2-2. CE16NGF Mid Grids Test Result Comparison

[ ]a,c

2.4 REFERENCES

2-1: WCAP-9401-P-A, "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," August 1981.

2-2: CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," August 1981.

2-3: APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report," May 2015.

2-4: NRC NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP) Section 4.2, "Fuel System Design," Revision 3, March 2007, Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces."

3 FUEL ASSEMBLY DYNAMIC CHARACTERISTICS AT EOL CONDITIONS

This section describes the test protocol for determining the fuel assembly dynamic characteristics at EOL conditions. The test protocol uses Westinghouse and CE PWR fuel assemblies with simulated EOL grids for determining fuel assembly dynamic characteristics at EOL conditions. A more detailed description is provided in the subsequent subsections.

The same test protocol has been previously applied to current Westinghouse and CE PWR fuel designs for BOL conditions.

The test protocols are described in NRC-approved TRs WCAP-9401-P-A (Reference 3-1) and CENPD-178-P, Rev. 1-P (Reference 3-2).

The main aspect of the testing described in this section that is different from what has been previously approved by the NRC for current Westinghouse and CE PWR fuel designs is as follows:

[ ] a,c

Even though this aspect has not been approved for current Westinghouse and CE PWR fuel designs, it has been approved by the NRC for the AP1000 plant as described in TR APP-GW-GLR-153, Rev. 1 (Reference 3-3).

For the testing performed in this program and described in this section, two fuel assembly designs were used: the Westinghouse RFA/RFA-2 and CE16NGF designs.

The results presented in this section are for the purpose of demonstrating the test protocol and for demonstrating the EOL effects to determine the fuel assembly dynamic characteristics at EOL conditions. The test protocol is applicable to all Westinghouse and CE PWR fuel designs.

3.1 EOL FUEL ASSEMBLY MECHANICAL TESTS

[ ] a,c

The mechanical tests obtained the fuel assembly (FA) static and dynamic characteristics, including FA modal frequencies and modal shapes, FA stiffness, FA structural damping, and fuel assembly impact forces.

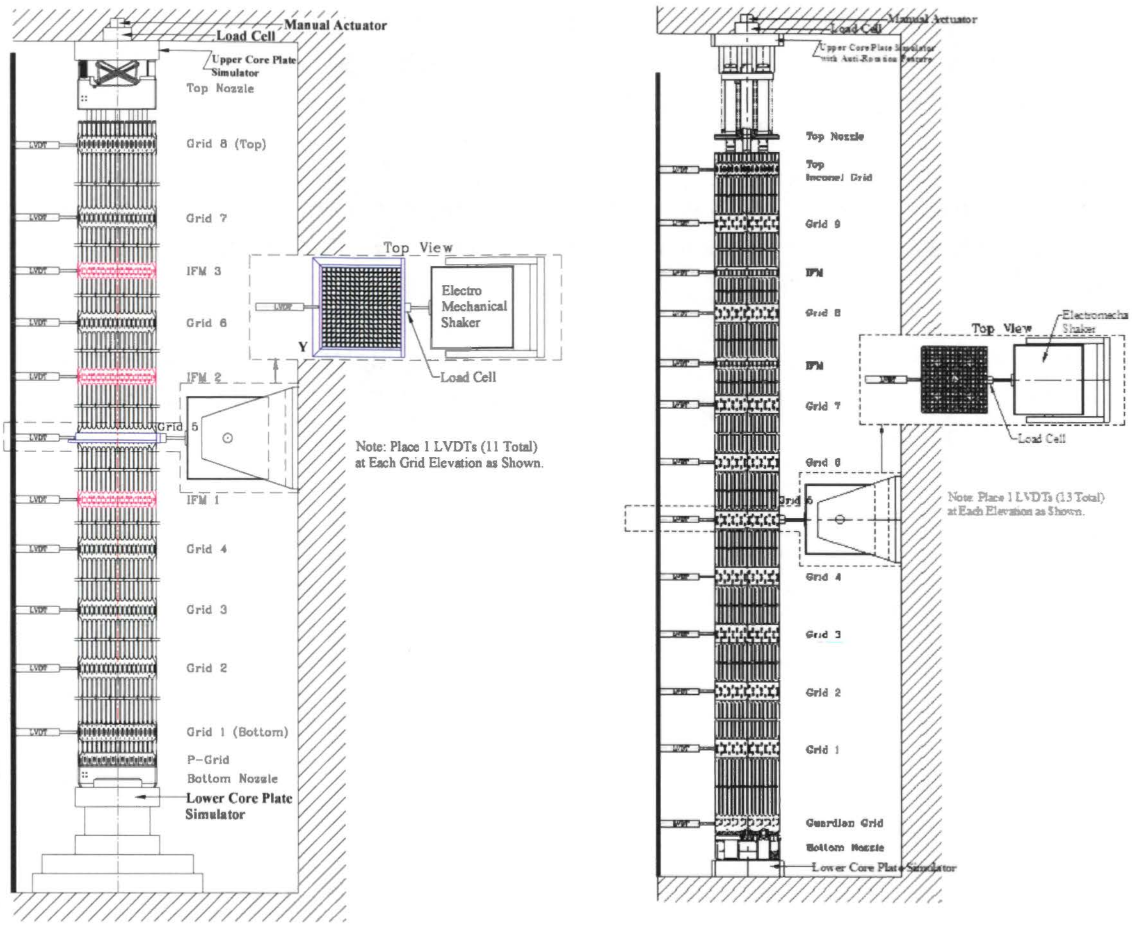
The FA lateral vibration tests were performed to obtain fuel assembly dominant modal frequencies and modal shapes. The test fuel assembly is held with nominal hold-down force in

the test stand with simulated lower and upper core plates. The typical mechanical test setup for the lateral vibration tests is shown in Figure 3-1. A shaker is used to provide a sinusoidal excitation force at approximately the middle of the test fuel assembly.

Two fuel assembly designs were tested. One test was performed for the RFA/RFA-2 fuel assembly design for Westinghouse 12-foot cores. This assembly design features a 17x17 array with a 0.374-inch diameter fuel rod. The RFA/RFA-2 design has six mid grids and three intermediate flow mixing (IFM) grids.

The other fuel assembly design that was tested is the CE16NGF fuel for CE cores. This assembly design features a 16x16 array with 0.374 inch diameter fuel rods. The CE16NGF design has nine mid grids and two IFM grids.

a,c



(a) 17RFA-2 Fuel Assembly (b) **CE16NGF** Fuel Assembly

Figure 3-1. Typical Fuel Assembly Lateral Vibration Test Setup

Table 3-1. Modal Frequencies of RFA/RFA-2 Fuel Assembly (at Room Temperature and in Air)

a,c

Table 3-2. Modal Frequencies of CE16NGF Fuel Assembly (at Zero-gap and EOL Conditions)

[ ] a,c

Note: Zero-gap results are from tests outside this PWROG program

3.2 REFERENCES

3-1: WCAP-9401-P-A, "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," August 1981.
3-2: CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," August 1981.
3-3: APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report," May 2015.

4 FUEL ASSEMBLY FLOWING WATER DAMPING

This section describes the test protocol for determining the fuel assembly flowing water damping ratio. The test protocol uses Westinghouse and CE PWR fuel assemblies with simulated EOL grids for determining the fuel assembly flowing water damping ratio at EOL conditions. A more detailed description is provided in the subsequent subsections.

Still water damping has been previously applied to current Westinghouse and CE PWR fuel designs as described in NRC-approved TRs WCAP-9401-P-A (Reference 4-1) and CENPD-178-P, Rev. 1-P (Reference 4-2)

The main aspects of the testing described in this section that are different from what has been previously approved by the NRC for current Westinghouse and CE PWR fuel designs are:

[ ] c

Even though there is no NRC-approved test protocol for determining the fuel assembly flowing water damping ratio for current Westinghouse and CE PWR fuel designs, the test protocol in this program is consistent with the test protocol approved by the NRC for the AP1000 plant as described in Reference 5 of Section H of TR APP-GW-GLR-153, Rev. 1 (Reference 4-3).

Even though crediting flowing water damping for EOL conditions has not been approved for current Westinghouse and CE PWR fuel designs, it has been approved by the NRC for the AP1000 plant as described in TR APP-GW-GLR-153, Rev. 1 (Reference 4-3).

[ ] c

For the testing performed in this program and described in this section, two fuel assembly designs were used: the Westinghouse RFA/RFA-2 and CE16NGF designs.

The results presented in this section are for the purpose of demonstrating the test protocol and for determining the flowing water damping ratio at EOL conditions. The test protocol is applicable to all Westinghouse and CE PWR fuel designs.

[ ] c

4.1 DESCRIPTION OF FLOWING WATER DAMPING TESTS

a,c

a,c

Figure 4-1. Test Loop Pressure Vessel and Pluck Mechanism

a,c

Figure 4-2. Flow Housing and Pressure Vessel (Top View)

4.2 BUNDLE FLOW RATE

The flow housing used in the flowing water damping tests was larger than the standard flow housing used for fuel assembly pressure drop tests, in order to accommodate the large fuel assembly lateral vibration. A portion of the flow bypassed the assembly and flowed along the sidewalls. Therefore, the flow through the fuel bundle could not be directly measured. The average bundle flowrate was calculated based on the measured fuel assembly lift force and the known bundle loss coefficients measured with the standard flow housing. The free body diagram of the test fuel assembly subjected to external forces in the test loop is shown in Figure 4-3.

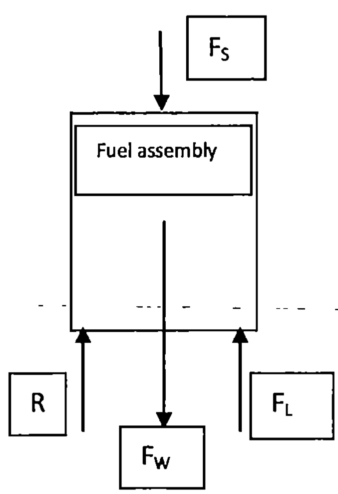


Figure 4-3. Fuel Assembly Free Body Diagram

In this diagram:

F_S – top nozzle spring hold-down force

F_W – fuel assembly weight included buoyancy force

F_L – the lift force due to flow impingement

R – reaction force on the lower core plate (measured by load cells)

Performing a force balance in axial direction:

$$\sum F_{axial} = 0$$

$$-F_S - F_W + R + F_L = 0 \tag{4-1}$$

Because F_W and F_S are constant for the same test temperature and without fuel assembly lift off, the lifting force is given by the change in the load cell readings. The average bundle flowrate is calculated based on the lift force and the known bundle loss coefficients.

a,c

Figure 4-4. RFA/RFA-2 Bundle Flow Rate Test Results

[ ] a,c

Figure 4-5. CE16NGF Bundle Flow Rate Test Results

4.3 FUEL ASSEMBLY FLOWING WATER DAMPING TEST CONDITIONS

[ ] a,c

4.4 FLOWING WATER DAMPING CALCULATION METHOD

Fuel assembly damping was obtained by pluck tests. Pluck testing is performed by displacing the middle of the assembly at an initial lateral displacement and releasing the assembly to allow a free vibration with no initial velocity. The pluck test, also known as the "decay method," is used to obtain the damping ratio of a damped dynamic system. The decay rate, a measure of damping, is expressed as the ratio of successive amplitudes. If x_i and x_{i+1} represent the amplitudes for the i[th] and (i+1)[th] successive cycle, the logarithm of the ratio of two successive cycles is called the logarithmic decrement (here called two successive amplitudes method, shown in Figure 4-6) and is denoted as (Reference 4-4):

$$\delta = \ln\frac{x_i}{x_{i+1}} = \frac{2\pi\zeta}{\sqrt{1-\zeta^2}} \qquad (4\text{-}2)$$

Solving for the damping ratio, ζ, results in the following:

$$\zeta = \frac{\delta}{\sqrt{4\pi^2 + \delta^2}} \tag{4-3}$$

The damping ratio obtained from the pluck test is based on the classic damping definition (Reference 4-4).

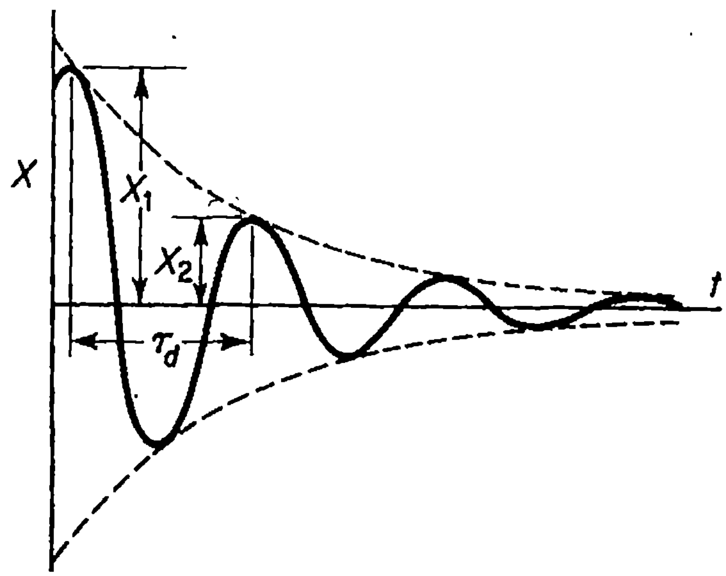


Figure 4-6. Illustration of Two Successive Amplitude Method

Fuel assembly damping ratios in air can be reasonably obtained by the two successive amplitudes method described in Equations (4-2) and (4-3) taken from Reference 4-4. However, the damping coefficients in flowing water are difficult to obtain by using Equations (4-2) and (4-3). As expected, fuel assembly damping in flowing water is much higher than in air. Figure 4-7 shows typical assembly displacement histories. Since the damping is so high and the assembly oscillation decays quickly, even the first vibration cycle is hard to recognize. To obtain accurate damping coefficients for high damping cases, the initial displacement and first response method based on classic vibration theory (References 4-4, 4-5, and 4-6) is therefore used.

a,c

Figure 4-7. Fuel Assembly Decay Motion in Flowing Water

The fuel assembly oscillatory motion after a quick release from the initial displacement can be expressed by a classic vibration equation ($\zeta < 1.0$, Reference 4-4):

$$x(t) = e^{-\zeta\omega_n t}\left(\frac{\dot{x}(0)+\zeta\omega_n x(0)}{\omega_n\sqrt{1-\zeta^2}}\sin\sqrt{1-\zeta^2}\,\omega_n t + x(0)\cos\sqrt{1-\zeta^2}\,\omega_n t\right) \quad (4\text{-}4)$$

Where:

$x(0),\ \dot{x}(0)$ – Initial displacement and velocity, respectively

ω_n – Natural frequency

For a pluck test with $\dot{x} = 0$, solving Equation (4-4) for the ratio $x(t)/x(0)$ gives

$$x(t)/x(0) = e^{-\zeta\omega_n t}\left(\frac{\zeta\omega_n}{\omega_n\sqrt{1-\zeta^2}}\sin\sqrt{1-\zeta^2}\,\omega_n t + \cos\sqrt{1-\zeta^2}\,\omega_n t\right) \quad (4\text{-}5)$$

where $x(0)$ is the initial pluck displacement and $x(t)$ is the response as a function of time.

When the damping coefficient is higher than 0.4, the oscillatory motion decays very quickly, shown in Figure 4-7. It is difficult to recognize a full oscillatory cycle. However, the pluck initial displacement and the first minimum amplitude can be measured with much better accuracy. Setting $x = x(\min)$ (first response peak), the vibration duration is then π (1/2 cycle); therefore,

$$\sqrt{1-\zeta^2}\,\omega_n t = \pi \quad (4\text{-}6)$$

and $\omega_n t = \dfrac{\pi}{\sqrt{1-\zeta^2}}$ (4-7)

Equation (4-5) with $x(\min)/x(0)$, becomes

$x(\min)/x(0) = e^{-\zeta\omega_n t}(0-1)$ or $-x(0)/x(\min) = e^{\zeta\omega_n t}$ (4-8)

Substituting Equation (4-7) into Equation (4-8), Equation (4-8) becomes

$$\delta_\pi = \ln\frac{x(0)}{-x(\min)} = \frac{\zeta\pi}{\sqrt{1-\zeta^2}} \quad (4\text{-}9)$$

$$\zeta = \frac{\delta_\pi}{\sqrt{\pi^2+\delta_\pi^2}} \quad (4\text{-}10)$$

Equation (4-10) is a special case of Equation (4-3), when the natural logarithm of the ratio of the initial displacement to the first half-cycle amplitude is used. The essential condition for using Equations (4-9) and (4-10) is the initial velocity is equal to zero. Equations (4-9) and (4-10) are used to obtain damping ratio from the pluck tests in this report.

4.5 FUEL ASSEMBLY FLOWING WATER DAMPING TEST RESULTS

a,c

a,c

Figure 4-8. RFA/RFA-2 Damping Ratios in Still and Flowing Water at 100°F

a,c

Figure 4-9. RFA/RFA-2 Damping Ratios in Still and Flowing Water at 200°F

a,c

Figure 4-10. RFA/RFA-2 Damping Ratios in Still and Flowing Water at 300°F

a,c

Figure 4-11. RFA/RFA-2 Damping Ratios in Still and Flowing Water at 380°F

4.6 FLOWING WATER DAMPING RATIO

a,c

Figure 4-12. RFA/RFA-2 Damping vs Bundle Velocity

a,c

Figure 4-13. CE16NGF Damping vs Bundle Velocity

a,c

Figure 4-14. RFA/RFA-2 Damping vs Density

a,c

Figure 4-15. RFA/RFA-2 Damping Ratio vs Bundle Velocity at 600°F

a,c

Figure 4-16. CE16NGF Damping Ratio vs Bundle Velocity at 600°F

4.7 BOUNDING DAMPING CURVE

To evaluate the effect of different PWR fuel designs on flowing water damping, the flowing water damping data from the previous test for the Westinghouse 19x19 fuel assembly (Reference 4-3) and RFA/RFA-2 are compared in Figure 4-17. The design features of the two test assemblies are summarized in Table 4-1.

a,c

Figure 4-17. Damping Ratio vs Bundle Velocity Curve Comparison.

Table 4-1. Comparison of Test Assembly Geometric Features

a,c

[ ]a,c

Fuel design specific flowing water damping coefficients at EOL conditions for Westinghouse and CE PWR fuel can be determined provided the test protocol described in this TR is used.

a,c

Figure 4-18. Bounding Damping Ratio vs Bundle Velocity Curve

4.8 FLOWING WATER DAMPING CREDIT WITH REACTOR COOLANT PUMP COASTDOWN DURING A SEISMIC EVENT

During a seismic event, reactor coolant pumps (RCPs) may trip, which would result in a pump coastdown and core flow reduction. When the flow rate decreases during the pump coastdown, the fuel assembly flowing water damping is also reduced. A conservative fuel assembly flowing water damping value was determined based on the flow rate during pump coastdown.

In the AP1000 EOL SSE (Safe Shutdown Earthquake) analysis, a conservative assumption was made that the loss of offsite power and RCPs trip occurred simultaneously with a seismic event. This assumption was approved by the NRC for analysis of the AP1000 plant (Reference 4-3).

The same assumption will be made for determining the appropriate flow rate for selecting the damping ratio to be used in the seismic analysis.

The following discussion provides an example for how to determine the flowing water damping ratio as a function of time during RCP coastdown. For a plant-specific seismic analysis, a plant specific RCP coastdown curve will be used.

Typical pump coastdown curves for Westinghouse 3-loop/4-loop of 12-foot cores and CE System 80 cores are very similar to that shown in Figure 4-19.

a,c

Figure 4-19. Typical RCS Pump Coastdown Curves

a,c

a,c

Figure 4-20. Typical 3-Loop RCS Pump Coastdown Curve

a,c

Figure 4-21. Damping Ratio vs. Coastdown Time for a Typical Westinghouse 3-Loop Unit

[ ] a.c

4.9 FLOWING WATER DAMPING CREDIT FOR A LOCA EVENT

[ ] a.c

4.10 REFERENCES

4-1: WCAP-9401-P-A, "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," August 1981.
4-2: CENPD-178-P, Rev. 1-P, "Structural Analysis of Fuel Assemblies for Seismic and Loss of Coolant Accident Loading," August 1981.
4-3: APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report," May 2015.
4-4: Theory of vibration with Applications, 3rd Edition, W. T. Thomson, Prentice Hall, 1988.
4-5: MUAP-13020-NP, "Axial Flow Damping Test of the Full Scale US-APWR Fuel Assembly," August 2013, Non-Proprietary Version, Mitsubishi Heavy Industries, Ltd, August 2013.
4-6: R. Y. Lu and D. D. Seel, Westinghouse USA, "PWR Fuel Assembly Damping Characteristics," Proceedings of ICONE 14, 14th International Conference on Nuclear Engineering, July 17-20, 2006, Miami, Florida, USA.
4-7: F. E. Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES).
4-8: S. Pisapia, et al. "Modal Testing and Identification of a PWR Fuel Assembly," Transactions of the 17th International Conference on Structural Mechanics in Reactor Technology (SMiRT 17), Paper #C01-4, Prague, Czech Republic, August 17-22, 2003.

5 CONCLUSIONS

To address NRC Information Notice (IN) 2012-09 for the Westinghouse and CE PWR fuel designs, the PWROG Analysis Committee and Westinghouse began work on this topical report. This topical report is based on the NRC-approved approach used for the AP1000 Core Reference Report and includes testing of Westinghouse and CE PWR fuel designs at simulated EOL conditions.

The topical report addresses the following three key items with respect to evaluating the structural integrity of fuel assemblies under faulted condition loads:

[ ]c

In this TR (PWROG-16043-P), the testing of two fuel designs is discussed. These were the Westinghouse 17x17 Robust Fuel Assembly-2, herein referred to as "RFA/RFA-2," and the CE CE16NGF fuel designs.

The test protocols for determining grid impact strength, dynamic characteristics of fuel assemblies, and flowing water damping described in this report characterize the EOL effects on fuel assemblies and that the test protocols can be applied to all Westinghouse and CE PWR fuel assembly designs.

This TR (PWROG-16043-P) does not supersede the NRC-approved TRs WCAP-9401-P-A and CENPD-178-P, Rev. 1-P. Following NRC approval of this TR, it will be used as the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant-specific seismic/LOCA analyses that will be performed in accordance with the current NRC-approved methods described in WCAP-9401-P-A and CENPD-178-P, Rev. 1-P.

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APPENDIX A

NRC Correspondence

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR TOPICAL REPORT PWROG-16043-P, REVISION 2, "PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09: 'IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH' FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS" PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

1.0 INTRODUCTION

By letter dated February 1, 2017 (Reference 1), the Pressurized Water Reactor (PWR) Owners Group (PWROG or the applicant), submitted to the U.S. Nuclear Regulatory Commission (NRC) staff for review licensing topical report (TR) PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" (Reference 2, henceforth referred to as the TR). Subsequent letters dated March 27, 2018, May 15, 2018, and May 15, 2018 (References 3, 4, and 5, respectively), provided additional information that supplemented the information provided in Reference 2. The TR is an extension of the previously approved methodologies described in WCAP-9401-P-A (Reference 6) and CENPD-178(P), Rev. 1-P (Reference 7), to assess the structural integrity of fuel assemblies under faulted condition loads. The methodologies described in the TR can be used to develop fuel assembly characteristics and damping coefficients for end-of-life (EOL) conditions that can then be used with the existing testing and analysis methodologies for seismic and loss-of-coolant accident (LOCA) events.

2.0 BACKGROUND

Seismic and LOCA events can result in external forces applied to the fuel assemblies (e.g., shaking and/or vibratory forces). Therefore, applicants must evaluate the fuel assembly structural response under these conditions to ensure that regulatory requirements are met with respect to control rod insertability and core coolability. In particular, the spacer grid performance is assessed to determine if plastic deformation is expected to occur, and the fuel assembly vibration behavior is quantified. Most PWR plants currently utilize the NRC approved testing and analysis methodologies described in References 6 and 7 for Westinghouse and CE fuel designs, respectively.

The NRC reviewed and approved References 6 and 7 based on the regulatory guidance provided in Appendix A to Chapter 4.2 of the Standard Review Plan (SRP or Reference 8). One assumption in the SRP Chapter 4.2 Appendix A guidance at the time, which is also in the current revision from 2007, is that beginning of life (BOL) is the time at which the crushing load for the spacer grids would be expected to be at a minimum. This assumption was based on the fact that irradiation tends to cause strengthening in metals and alloys in addition to embrittlement. Other effects that arise due to use in a reactor may include growth, cladding creep, and corrosion. The increase in strength was expected to more than offset the other effects associated with irradiated grids. Since applicants typically verify that the maximum load experienced by the spacer grids during LOCA and seismic events will not exceed the crushing load, use of BOL characteristics was considered to be conservative.

Enclosure

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Operating experience that came to light in the mid-2000s led the NRC staff to question the assumption that the spacer grid structural performance during LOCA and seismic events would not degrade significantly as a result of irradiation. The NRC subsequently issued Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" (Reference 9). This IN lists several factors that can affect the structural strength of the spacer grids, and singles out spacer grid spring relaxation as one that can have a significant effect on the fuel assembly mechanical characteristics and the spacer grid strength. While no specific action or response was required as a result of the IN, the NRC indicated that recipients would be expected to review the information for applicability and consider appropriate action to avoid similar problems.

This TR is the applicant's proposed approach to generically address the issue identified in the IN for licensees that use Westinghouse or CE fuel. In essence, this TR describes how to extend the testing and analysis methodologies in References 6 and 7 to determine an appropriate crushing load for spacer grids at EOL. In addition, the TR proposes a methodology that can be used to develop flowing water damping ratios that can then be credited in the LOCA and seismic analyses in a similar manner to the NRC approved still water damping ratios (as described in References 6 and 7). This provides a means for licensees to recover margin lost due to the effect of spacer grid spring relaxation on the fuel assembly mechanical characteristics.

In summary, the existing NRC approved testing and analysis methodologies will continue to be used, with all previously established limitations and conditions, but this TR extends the applicability of the relevant aspects of these methodologies to the extent necessary to address potential fuel assembly structural performance issues as a result of irradiation.

3.0 REGULATORY EVALUATION

Title 10, "Energy," of the *U.S. Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," contains requirements for the emergency core cooling system (ECCS) at commercial power plants. In particular, 10 CFR 50.46(b)(4) requires that "[c]alculated changes in core geometry shall be such that the core remains amenable to cooling." Any failure in the structural integrity of the fuel assemblies will typically change the core geometry, and the possibility needs to be evaluated.

The regulation at 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor design," states that "[t]he reactor core. . shall be designed with appropriate margin to assure that specified fuel design limits are not exceeded during...anticipated operational occurrences." Within the context of seismic events, this is implicitly addressed by ensuring adequate core coolability.

The regulation at 10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," states that "[t]he reactivity control systems shall be designed to... reliably [control] reactivity changes..." One of the primary reactivity control systems at current WEC and CE PWR plants is the rapid insertion of control rods to add sufficient negative reactivity to shut down the reactor. Reliable operation of this reactivity control system is conditional on the capability to insert the control rods. Vibrations or structural deformations may impede the control rod movement, and need to be evaluated.

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The regulation at 10 CFR Part 50, Appendix A, GDC 35, "Emergency core cooling," restates the requirement to maintain adequate emergency core cooling capability, which can be affected by the core geometry as discussed in 10 CFR 50.46(b)(4) (see above).

The regulation at 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," requires safety-related structures, systems, and components (SSCs), including reactor fuel, to be designed to withstand natural phenomena (such as earthquakes) without a loss of capability to perform safety functions. This GDC also requires consideration of "appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena." For example, a LOCA may be caused by a seismic event, so consideration of the effects from a combination of these two events may be appropriate.

Appendix S of 10 CFR Part 50 provides additional guidance for seismic events, and defines the Safe Shutdown Earthquake (SSE), Operating Basis Earthquake (OBE), and safety requirements for relevant SSCs. In general, stress, strain, and/or deformation limits should be defined for each SSC to ensure its functional capabilities during each event indicated by the regulatory requirements (typically OBE, LOCA+SSE, and SSE-only, though other combinations may be considered). These requirements are not explicitly addressed by the methodologies submitted for NRC review, however, the overall methodology that PWROG-16043 will supplement is intended to demonstrate that these requirements are met. Therefore, the NRC staff considered the potential impact of PWROG-16043 on how the 10 CFR Part 50 Appendix S requirements would be met.

The acceptance criteria for the structural response of fuel assemblies to externally applied forces, in order to satisfy the above criteria, are defined in Section 4.2, Appendix A of the SRP, otherwise known as NUREG-0800 (Reference 8). In general, the primary criteria are related to ensuring that core coolability and control rod insertability are maintained.

This TR is an application of an evaluation model to perform licensing analyses for an accident that the evaluation model has not previously been approved. As such, additional guidance for the evaluation may be found in SRP Chapter 15.0.2, "Review of Transient and Accident Analysis Methods" (Reference 10). This chapter includes provisions for the review of submittals related to evaluation models.

In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 along with the applicable acceptance criteria in SRP Chapters 4.2 Appendix A in conducting its review of the TR. Since the TR is effectively a supplement to existing methodologies, the scope of the NRC staff review was limited to the elements of the TR that represented a novel approach relative to the existing methodologies, and to verify the applicability of the existing methodologies when conducting tests and evaluations as described in the TR.

4.0 <u>TECHNICAL EVALUATION</u>

The intent of the TR is to avoid extensive modification of previously approved analysis methodologies documented in References 6 and 7 by focusing solely on the specific parameters that would be impacted by the EOL issues identified in IN 2012-09 (Reference 9). As such, the TR narrowly focuses on three primary parameters:

1. The allowable grid impact strength [
   ]

2. The fuel assembly modal frequencies [

] and

2. The fuel assembly flowing water damping ratio, [
]

As a result, some of the areas from SRP Chapter 15.0.2 are not applicable. In particular, the analysis methodologies described in References 6 and 7 are not being modified, only the empirical determination of key input parameters. Therefore, the accident scenario description, the phenomena identification and ranking, and code assessment from the previously approved methodologies remain valid. The NRC staff review of the TR focused on two of the specific areas described in SRP Chapter 15.0.2, as described below:

1. Evaluation methodology – the proposed testing and data analysis methodologies, including any potential limitations to their applicability.

2. Uncertainty analysis – the applicant's evaluation and propagation of uncertainties in the analysis of test data to obtain recommended values for the key parameters.

In addition, the NRC staff considered whether the applicant provided adequate quality assurance (QA) and documentation support for the proposed methodologies. This aspect is not explicitly discussed in detail for this safety evaluation (SE) because the documentation of the proposed methodologies are captured by the documents reviewed by the NRC during an audit dated October 17, 2017 (Reference 11) and that were found to have been appropriately summarized or otherwise characterized in the TR. The testing was performed under the auspices of the same QA program used to perform the testing for the previously approved methodologies to determine the key parameters for BOL grids and still water damping, which is acceptable. As such, the NRC staff acceptance of the adequacy of the applicant's evaluation methodologies and uncertainty analyses implicitly includes acceptance of the applicant documentation associated with that area.

4.1 EOL Grid Simulation

All of the proposed methodologies in the TR are based on a specific characterization of the impact of irradiation on the spacer grids. SRP Chapter 4.2 Appendix A (Reference 8) cites several possible irradiation-related effects relevant to spacer grids, and concludes that the combined impact would not be expected to lead to a more conservative result. This logic rests mainly on the fact that the significant increase in yield strength for the spacer grid material will more than offset the relatively minor effects from the remaining effects. As described in IN 2012-09 (Reference 9), operating experience has shown that spacer grid spring relaxation can have a significant adverse effect on spacer grid strength and fuel assembly mechanical characteristics. [

] Other than grid spring relaxation, the basic assessment in SRP Chapter 4.2 Appendix A that irradiation-related effects are bounded by the increase in the yield strength of the spacer grid material continues to be applicable. [

]

[

As discussed in the previous paragraph, the NRC staff found that the focus on the grid spring relaxation phenomenon as the key driver for the non-conservative behavior identified in spacer grids at EOL relative to BOL is appropriate. However, the material and geometry impacts of the thermal relaxation process must be reasonably similar to the irradiation-induced impacts that are being simulated.

[

] Therefore, the NRC staff requested additional information from the applicant regarding the thermal relaxation procedure used to produce the simulated EOL grids. [

] The applicant's response also confirmed that the material structural characteristics of the simulated EOL grids are the same, or slightly conservative, relative to the BOL grids.

[

] There are some situations where a spacer grid is exposed to a strongly non-uniform neutron flux, such as fuel assembly loading locations at or near the core periphery The NRC staff asked the applicant to address the potential impact on the grid failure mechanism due to non-random gradients in gap size that may be correlated with steep neutron flux gradients. [

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]

Finally, Chapter 2.1 of the TR described how the target average gap size was determined for a given spacer grid. [

]

Inadequate information was given in the TR to define the area of applicability for extrapolation of a given set of PIE data to the general population of EOL grid spacers of the same design, so the NRC staff requested that the applicant characterize how PIE data sets are generally defined in order to achieve their intended purpose.

The applicant responded in Reference 4 with an explanation of the statistical methodology underlying their determination of a target gap size for the simulated EOL grids. [

] this is a reasonably conservative approach to ensure that the average gap sizes for the simulated EOL grids will bound the average gap sizes for irradiated grids.

[

] The NRC staff agrees, however, the applicant did not describe how the rod burnups associated with the PIE measurements would be used to define the area of applicability for fuel assemblies qualified under this methodology. In a separate RAI response (RAI-2, documented in Reference 4), the applicant provided information that shows that the variation in gap sizes for varying burnups near EOL can be expected to be minor relative to the inherent randomness in gap sizes within a grid. In addition, the NRC staff noted that the methodology described in Reference 7 for testing of CE design fuel assemblies includes modeling for both BOL and EOL grids. [

] Consistent with this assessment, the results from the testing discussed in Sections 4.2 and 4.3 of this SE show [

] Therefore, any variations in burnup for the fuel assemblies used to obtain PIE measurements relative to the overall population of fuel assemblies being qualified under this methodology would not result in a significant difference in average gap size, certainly, much less than the inherent conservatism in the margin between the average measured gap sizes and the target gap size for the simulated EOL grids.

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[

] As a result, the NRC staff found the proposed approach to generate simulated EOL grids for use in testing in lieu of irradiated grids to be acceptable.

4.2 Spacer Grid Impact Strength

Chapters 2.2 and 2.3 of the subject TR discuss the application of the approved testing and data analysis methodologies from References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. In all respects, the testing and data analysis applications were consistent with References 6 and 7, [

]. The NRC staff understanding of the approval request from the applicant is that this change in criterion was adopted merely for demonstration purposes, not being submitted as an update to the Reference 6 methodology. In response to a RAI from the NRC staff (Reference 3), the applicant confirmed that this was the case. Therefore, this application was judged to be acceptable solely for the purpose of providing a more consistent basis for comparing the change in P(crit) for Westinghouse and CE fuel designs.

The simulated EOL grids contain [

] The NRC staff verified by inspection of the applicant's test documentation that the failure mechanism for the simulated EOL grids was the same as that for the BOL grids. Therefore, [

] As discussed in Section 4.2 of this SE, [

]

The NRC staff verified that the previously approved testing and data analysis methodologies from References 6 and 7 were appropriately applied to the simulated EOL grids. In addition, the NRC staff found reasonable assurance exists that the aforementioned methodologies remain applicable to the geometry of the simulated EOL grids. Therefore, the NRC staff found the methodologies to determine P(crit) to be acceptable for use in analysis of the simulated EOL grids.

4.3 Fuel Assembly Mechanical Characteristics

Chapter 3 of the TR discusses the application of the approved testing and data analysis methodologies from References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. The TR states that "[t]he same test protocol has been previously applied

to current Westinghouse and CE PWR fuel designs for BOL conditions," and that "[t]he test protocols are described in NRC-approved TRs…" with a citation to References 6 and 7. Therefore, the TR clearly characterizes the testing procedure for the simulated EOL grids to be identical to the previously approved testing procedure described in References 6 and 7, with the exception that the grids are simulated EOL grids as discussed in Section 4.1 of this SE.

The testing methodologies described in References 6 and 7 are primarily tests conducted on the structural members of the fuel assembly and the spacer grids, with no tests directly impacting the fuel rods. At BOL, the grid springs exert a frictional force on the fuel rods, so the spacer grids and fuel rods are mechanically coupled to some extent. During the fuel assembly vibration tests, the fuel rods contribute to the fuel assembly mechanical performance by virtue of this mechanical coupling. [ ]

[ ]

4.4 Procedure to Determine Flowing Water Damping Ratios

Chapter 4 of the TR describes a methodology to determine fuel assembly flowing water damping ratios and apply them in lieu of previously approved still water damping ratios to characterize the fuel assembly mechanical behavior during seismic and LOCA events. Since the damping ratio due to flowing water is expected to be higher than that for still water, this approach could help recapture margin lost due to the impact of grid spacer relaxation on the fuel assembly stiffness. [ ]

Chapters 4.1 through 4.3 describe the test apparatus and data collection performed to support an empirical determination of the flowing water damping ratios. [ ] However, the hydraulic characteristics for the fuel assembly are well characterized based on prior testing. [ ] Since the loss coefficients for the fuel assembly designs have been approved by the NRC for use in other

analyses and would not be expected to vary significantly as a result of the use of simulated EOL grids, this approach for determining flow velocities through the fuel assembly is acceptable.

The existing analysis methodologies, most notably the Reference 7 methodology for CE fuel, [ ] Testing performed on similar fuel assembly designs using a range of different approaches, as documented in References 14 and 15, yield consistent results. [ ] This shows that the proposed methodology yields results consistent with currently approved methodologies.

The flowing water damping ratio correlation was developed based [ ] Therefore, there will be no inconsistency in the application of damping ratios for fuel assemblies at different burnup conditions.

Based on the data collected from the tests, a damping ratio was determined for each test based on classical vibration theory. [ ] Chapter 4.5 of the TR presents results from the tests. One of the most important conclusions that can be observed directly from the test results is that [ ] Since the use of lower damping ratios in developing the correlation is conservative, this was an acceptable choice to make.

Chapter 4.6 of the TR discusses the data analysis approach used to determine bounding correlations for each fuel assembly design. This approach can be summarized thus: [ ] The overall approach appears to capture the relevant dependencies, however, there is no propagation of the uncertainties due to scatter in data through the steps noted above. [

]

The applicant responded in Reference 5 with information indicating that the fitting approach used to determine the bounding curve was fundamentally a best estimate approach to derive the 600 °F curve based on the selected data set. [

]

[

]

Finally, Chapter 4.7 proposes use of a flowing water damping ratio correlation based on the [ ] fuel assembly design as a generically bounding correlation that may be used with any fuel assembly design without further justification. The methodology discussed above may be used to develop fuel assembly design specific correlations, but the [ ] correlation is proposed for use as a bounding curve for all Westinghouse and CE fuel designs. The justification provided is that the [ ] fuel assembly design proposed for the [ ] reference plant contains a number of significant design differences, but test results show that the flowing water damping ratio is very similar to the [ ] fuel. The CE fuel design tested had [

] This behavior is bounded by the [ ] correlation, so this is acceptable. However, [

] Therefore, the similarity in results is not surprising.

In order to establish that the proposed correlation can be used as a generic bounding curve, its applicability must be limited to spacer grids with very similar geometry characteristics. This is accomplished via a condition to the TR. Information submitted in References 14 and 15 provide information for other PWR fuel assembly designs that suggests that, in fact, the [

]

As long as the geometry characteristics of the spacer grids associated with a different fuel assembly do not differ significantly from the [ ] spacer grid, the NRC staff finds that reasonable assurance exists that other fuel assembly designs will have flowing water damping ratios near or above the proposed bounding curve. The proposed application includes use of a minimum value for the analysis duration rather than a more realistic average value, which incorporates some additional conservatism that offsets the potential for slightly lower flowing water damping ratios for some fuel assembly designs relative to the proposed bounding curve.

Based on the information provided in the TR, as supplemented by responses to requests for additional information from the NRC staff, the testing protocol and data analysis methodologies described to determine appropriate flowing water damping ratios were determined to be appropriate for their intended purpose. In addition, [ ] This latter condition was captured in Section 5.0.

4.5 Analytical Application of the Flowing Water Damping Ratios

Chapters 4.8 and 4.9 of the TR describe when and how the flowing water damping ratios can be utilized in seismic and LOCA analyses, respectively. The primary parameter used to establish the appropriate value for the flowing water damping ratio is the fluid velocity through the fuel assembly. For a given plant, this parameter is directly correlated with the core flow. Therefore, the discussion in the TR primarily focuses on the characterization of a bounding core flow for any given time of interest during the event being analyzed. Once an appropriate value is determined, then plant-specific information can be used to establish an appropriate flow velocity to use with the flowing water damping ratio correlation. [ ] In general, since lower flow velocities result in lower flowing water damping ratios, any factor that may lead to a reduction in the core flow rate will provide more conservative results. For a given analysis, [ ]

For the seismic analysis, two key assumptions are made to minimize the total core flow. First, [ ]

Secondly, [

] At this time, the flowing water damping ratio will be at a minimum, and lower than the average flowing water damping ratio for the interval. Since these assumptions both act to minimize the flowing water damping ratio, they are conservative.

For the LOCA analysis, the core flow rates are to be obtained directly from the LOCA analyses, as long as axial flow is maintained. [

As a result, the NRC staff finds that the LOCA analysis conditions are an acceptable source for a bounding core flow rate for the purpose of determining flowing water damping ratios.

A second limitation of the flowing water damping ratios is that the data used as a basis for the correlation were based on single phase liquid flow through a fuel assembly. The conditions under which the flowing water damping ratios are expected to be credited—seismic events and the first ~1 second of a LOCA event—are not expected to involve two phase flow in the core. However, the TR does not explicitly limit the use of flowing water damping ratios to single phase flow conditions, so a limitation was included in Section 5.0 to ensure that, if this methodology is applied to conditions that deviate from expectations, the correlation will not be used outside the bounds of its applicability.

The NRC approval of Reference 13 included review of information demonstrating that the Westinghouse models were capable of capturing the dynamic behavior of fuel assemblies for pluck response inside a flow loop, for the vibration range of interest Since the flowing water damping ratios are very similar for the RFA/RFA-2 curves being proposed for use as a bounding curve for all fuel assembly designs and the Reference 13 fuel design contained a similar spacer grid design, this finding is applicable to the subject LTR as well. However, without further validation, the dynamic models cannot be assumed to maintain reasonable accuracy for damping ratios that go significantly beyond the current area of applicability. Therefore, any use of damping ratios significantly higher than the proposed bounding curve must be supported by validation against test data that demonstrates that the analytical models remain valid for the higher damping regime. A limitation was included in Section 5.0 to ensure that this potential limitation of the analytical models is addressed, if necessary.

The guidance provided in the TR to credit flowing water damping in seismic and LOCA analysis was reviewed by the NRC staff and determined to produce acceptably conservative results for the expected analysis conditions. Therefore, the NRC staff finds the proposed application of flowing water damping credit for evaluation of fuel assembly mechanical behavior during seismic and LOCA events to be acceptable.

4.6 Known Legacy Issues

There are a number of potential issues with the previously approved methodologies described in References 6 and 7. They include:

- [

]

These issues may have been addressed for legacy fuel assembly designs based on expected fuel assembly grid behavior and testing. However, the current approved methodologies do not provide a generic approach to do so. Therefore, the assumptions inherent in the technical justification for these issues need to be evaluated on a case-by-case basis for new fuel assembly designs, which may depend on consideration of all attributes of the proposed revisions to the plant licensing basis The new proposed approach to credit flowing water damping ratios represents a more realistic approach. As such, there is a reduction in conservatism for this approach relative to the previously approved approach to credit still water damping. Therefore, the overall justification for the above issues must be re-evaluated to ensure that the overall analysis remains conservative.

As discussed above, the NRC staff identified some technical issues that are not explicitly addressed by the currently approved methodology. They may have been addressed for current fuel assembly designs, however, the use of a more realistic flowing water damping ratio represents a reduction in conservatism for the damping ratio approach relative to the previously approved approach. Therefore, the NRC staff is imposing limitations and conditions to ensure that the overall conservatism of the analysis is acceptable

5.0 <u>LIMITATIONS AND CONDITIONS</u>

Some limitations and conditions are necessary to ensure that the methodology discussed in the TR is limited to the applications for which it is valid. These limitations and conditions are listed below.

1. [

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6.0 CONCLUSIONS

In the TR, the applicant presented new models and methods to extend the applicability of existing methodologies to evaluate spacer grid and fuel assembly mechanical behavior during seismic and LOCA events. The following conclusions are provided here in summary as they apply to licensees who may want to adopt the methodologies described in the TR with existing methodologies in References 6 and 7 to address the effect of irradiation on the mechanical properties of fuel assemblies.

Since the TR is not proposing any change to the previously approved testing and analysis methodologies for seismic and LOCA events, the NRC staff performed a graded review of the methodologies that took into consideration the fact that most aspects of the testing and analysis have already been addressed as part of prior NRC reviews. The applicant requested approval for three distinct enhancements to their existing methods: (1) use of simulated EOL spacer grids to assess spacer grid crush strength at EOL; (2) use of simulated EOL spacer grids to assess fuel assembly mechanical characteristics, such as stiffness, at EOL; and (3) use of a new methodology to develop flowing water damping ratios that can be used in lieu of the currently approved still water damping ratios

The NRC staff examined the proposed approach to produce simulated EOL spacer grids and use them with previously approved methodologies, and determined that the simulated EOL spacer grids would adequately capture the non-conservative impacts due to irradiation. The staff also determined that the [ ] The NRC staff's findings were based primarily on the specific material type (zirconium alloy) and general grid design covered by the information presented in the TR, [ ]

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The use of flowing water damping ratios is not an entirely new approach to develop more realistic parameters that help mitigate the impact of vibratory loads, because it is similar to a methodology submitted as part of the NRC approval of the AP1000 reference plant design (Reference 14). However, this is the first time that it is being applied more generically to Westinghouse and CE fuel. In particular, the applicant is proposing the use of a bounding curve that is applicable to all spacer grids used in Westinghouse and CE fuel, along with a general methodology that can be used to generate fuel design specific curves. The staff reviewed the information submitted in the TR along with responses to requests for additional information, and determined that the methodology was appropriate for both purposes. Additionally, the guidance provided for utilization of flowing water damping ratios in seismic and LOCA analyses was found to be appropriate for their intended use, with the limitations that: (1) the flowing water damping ratios are only valid for single phase liquid flow, and (2) the dynamic models used to predict the fuel assembly response under vibratory and damping loads must be verified to remain reasonably accurate for higher damping regimes by validation against test data, prior to use for safety analysis purposes.

The NRC staff also acknowledged some legacy issues with lack of clear guidance to address certain aspects of current NRC regulations. Since approval of use of specific fuel assembly designs at specific plants may have depended on consideration of fuel design specific characteristics that would disposition or offset the legacy issues, the NRC staff finds that any reduction in analytical conservatism should not be made without addressing these legacy issues, as discussed in Section 4.6. The use of flowing water damping ratios represents one such reduction in analytical conservatism, therefore, a condition for use of the new damping ratios is that the legacy issues need to be addressed.

In summary, the NRC staff finds that the information provided in the TR and responses to NRC staff RAIs adequately demonstrates that the proposed methodologies to address EOL effects on spacer grids and to recover margin through credit for flowing water damping are acceptable for use with existing methodologies that the NRC has previously found to be acceptable for analysis of fuel assembly structural behavior during seismic and LOCA events. The NRC staff approval of these methodology extends to all Westinghouse and CE fuel designs, contingent on adherence to the limitations and conditions set forth in Section 5.0.

7.0 REFERENCES

1. PWROG letter OG-17-12, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Submittal of PWROG-16043-P, Revision 2, 'PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169R2," February 1, 2017 (ADAMS Accession No. ML17039B050)

2. PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 2017 (ADAMS Package Accession No. ML17039B061)

3. PWROG letter OG-18-62, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request

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for Additional Information, RAIs 4 and 5 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," March 27, 2018 (ADAMS Accession No. ML18100A053)

4. PWROG letter OG-18-104, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re "Transmittal of the Response to Request for Additional Information, RAIs 1, 2, and 3 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," May 15, 2018 (ADAMS Accession No. ML18143B462)

5 PWROG letter OG-18-105, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re. "Transmittal of the Response to Request for Additional Information, RAI 6 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," May 15, 2018 (ADAMS Accession No. ML18144A760)

6 WCAP-9401-P-A, Revision 0, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," September 1981 (ADAMS Accession No. ML090280466 (Non-Publicly Available))

7. CENPD-178(P), Revision 1-P, "Structural Analysis of Fuel Assemblies for Seismic & LOCA Loading," August 1981 (ADAMS Accession No. ML14122A086 (Non-Publicly Available))

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 4.2, Revision 3, "Fuel System Design," March 2007 (ADAMS Accession No ML070740002)

9. NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," dated June 28, 2012 (ADAMS Accession No ML113470490)

10. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 15.0.2, Revision 0, "Review of Transient and Accident Analysis Methods," March 2007 (ADAMS Accession No ML070820123)

11. NRC letter from Brian Benney, Senior Project Manager, Licensing Processes Branch, Division of Policy and Rulemaking, USNRC, to Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, re "Summary Report for the October 17, 2017, Audit in Support of the Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09. 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 8, 2018 (ADAMS Accession No. ML17326A003)

12 Framatome ANP, Inc. letter NRC:03:051, James F. Mallay, Director, Regulatory Affairs, Framatome ANP, Inc., to USNRC document control desk, re. "Closure of Interim

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Report 02-002, 'Spacer Grid Crush Strength – Effects of Irradiation,'" August 8, 2003 (ADAMS Accession No ML032240425)

13 WCAP-17524-P/NP-A, Revision 1, "AP1000 Core Reference Report," May 2015 (ADAMS Accession No. ML15180A175)

14. Westinghouse letter LTR-NRC-13-26, James A. Greshman, Manager, Regulatory Compliance, Westinghouse Electric Company, to USNRC document control desk, re: "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor (Proprietary/Non-Proprietary)," April 30, 2013 (ADAMS Accession No. ML13128A017)

15. Framatome Inc. report ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018 (ADAMS Package Accession No ML18144A816)

Principal Contributor: Scott Krepel, NRR/DSS/SNPB

Date: August 22, 2018

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Comment Number	Text Location		PWROG Comment (paraphrased)	NRC Response
	Page	Line		
1	Multiple	Multiple	Some phrases (e g., page 1, lines 17-18) are inconsistent with how the purpose of the TR is characterized in Section 1.3. Throughout the draft safety evaluation (DSE), the term "methodology" should be replaced with alternative terms to clarify the relationship between the analytical methods and the test protocol or approaches used to develop parameters for use in the analytical methods.	The staff agrees that the distinction is useful to support consistency and clarity in the discussion, and has generally made changes to the DSE consistent with what PWROG recommended. Note this comment and response encompasses all proposed changes that are not explicitly identified in the following comments
2	2	48	Clarification that statement applies to both LOCA and seismic events.	The staff agrees, and the proposed changes were incorporated as-is.
3	3	21-30	The paragraph associated with 10 CFR 50, Appendix S should be deleted, since the specific criteria are discussed in the following paragraph.	The staff does not agree The regulations define the requirements, while the criteria provided in the SRP and other guidance documents are not binding. As such, the regulations form the regulatory basis, while the SRP provides additional guidance for acceptable approaches to demonstrate that the regulatory requirements are met. The staff did revise the paragraph slightly to refer to "criteria" rather than "limits" to be consistent with the discussion elsewhere in the DSE.
4	3	32-37	Editorial changes proposed for readability.	The staff agrees, and the proposed changes were incorporated as-is.
5	3 4	40-49 1-2	These paragraphs are not consistent	The staff agrees, consistent with the response to comment #1

Comment Number	Text Location		PWROG Comment (paraphrased)	NRC Response
	Page	Line		
			with the intent of the TR to provide an alternate approach for determining input, as opposed to a change in the analysis method.	(above) The paragraphs were deleted, however, some additional text was added to clarify that the NRC staff did consider the applicability of the analysis methodologies described in References 6 and 7 when the parameters of interest are developed with the new approaches. When prior review and approval of analytical methods are based, in part, on recommendations for development of input parameters, this aspect cannot be completely divorced from the analytical methods.
6	6 & elsewhere	22 & elsewhere	Replace use of the word "Chapter" with "Section."	The staff agrees, and the proposed changes were incorporated as-is.
7	7	34-46	Proposed rewrite for clarity	The staff agrees, and the proposed changes were mostly incorporated as-is. However, the characterization of the plastic deformation demonstration for Westinghouse grids as an "exception" was left in, since this is an important clarification—this is not consistent with References 6 and 7, however, PWROG is not requesting approval for use of this approach
8	9	44	Update text to cite specific references-- for approved methods.	The staff agrees that this edit provides additional clarity, but additional detail was included for completeness.
9	10	28	Proposed replacement of text.	The staff disagrees. The proposed rewrite would change the meaning of the sentence and be inconsistent with the discussion in the following two paragraphs.
10	12	30	Proposed rewrite to be consistent with the TR.	The staff agrees, and the proposed changes were incorporated as-is.

Comment Number	Text Location		PWROG Comment (paraphrased)	NRC Response
	Page	Line		
11	13 14	4-15 42-47	The issue with validity of the models used to predict the dynamic behavior of fuel assemblies pertain to the analytical methods, which are not being updated or reviewed by the NRC. Therefore, it is inappropriate to address this issue as part of the review of PWROG-16043.	The staff disagrees. While the analytical methods are not being updated, the proposed approach may produce damping ratios that are much higher than the range considered when the analytical methods were reviewed Consequently, questions exist about the applicability of the analytical methods for much higher damping ratios. The staff revised the limitation and condition to provide more latitude in what kind of information a licensee must provide in order to credit significantly higher damping ratios
12	13 14	23-49 2-18 36-40	The issues discussed regarding legacy issues pertain to the analytical methods, which are not being updated or reviewed by the NRC. Therefore, it is inappropriate to address this issue as part of the review of PWROG-16043.	The staff agrees that the legacy issues are not part of the review scope for this TR, but disagrees that they cannot be considered as part of the basis for approval of this TR. The legacy issues were not considered as part of the review of PWROG-16043. However, PWROG-16043 requests approval for an approach that removes conservatism from analyses performed using the analytical method. This conservatism, among other factors, was used to risk inform the staff's decision not to pursue resolution of the legacy issues As a result, if licensees wish to remove this conservatism, they need to provide information to resolve the issues. This information has already been provided and reviewed in some cases (e.g , Reference 13). The staff rewrote Section 4.6 and the limitation and condition to provide better clarity on the issues that need to be

Comment Number	Text Location		PWROG Comment (paraphrased)	NRC Response
	Page	Line		
				addressed and how they can be addressed
13	15	17-22	Proposed rewrite to be more consistent with the TR.	The staff agrees, and the proposed changes were incorporated as-is
14	15	28-29	Proposed rewrite to be more consistent with discussion elsewhere in the draft safety evaluation.	The staff agrees, and the proposed changes were incorporated as-is
15	15	37-39	Proposed rewrite for clarity and change reference to point to approved TR instead of RAI response	The staff agrees, and the proposed changes were incorporated as-is
16	15 16	47-49 1-11	Proposed deletion consistent with comments 11 and 12, above	The staff disagrees; see responses to comments 11 and 12, above. The text was revised to be consistent with disposition of these comments

PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

PWROG-16043-P, Revision 2
Docket Number 99902037

January 16, 2019

OG-19-13

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
PWROG Comments on the NRC Draft Safety Evaluation for PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength for Westinghouse and CE PWR Fuel Designs" (PA-ASC-1169)

Reference:

1. Draft Safety Evaluations by the Office of Nuclear Reactor Regulation for topical report PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength for Westinghouse and CE PWR Fuel Designs" Pressurized Water Reactor Owners Group (PWROG), (ML18186A634), dated August 22, 2018.

At the October 17, 2018 meeting between the PWROG and NRC to discuss the major comments on the NRC Draft Safety Evaluation (DSE) for PWROG-16043, the PWROG agreed to provide formal comments on the NRC DSE for PWROG-16043-P.

The PWROG has the following major comment on the Draft Safety Evaluation (DSE):

Section 4.6 of the DSE discusses "known" legacy issues associated with NRC approved NSSS vendor analytical methods that are unrelated to the purpose of this Topical Report (TR).

The discussion of legacy issues also includes concerns associated with the NRC approved NSSS vendor analytical methods that are used for beginning of life (BOL) conditions. If there are concerns with these methods at BOL, they would apply to all licensees who used the methods to support their current licensing basis. The TR only addresses the end of life (EOL) conditions, and only the PWROG members that funded the project, have access to it. The PWROG is the applicant requesting approval of the TR, not the NSSS vendor. In addition, as discussed in the TR, the NSSS vendor analytical methods are not being revised as part of the PWROG program documented in the TR. Therefore, the Safety Evaluation for the TR is not the appropriate vehicle to communicate to an NSSS vendor any potential issues associated with NRC approved analytical methods.

Addressing legacy issues is outside the scope of the TR review and the PWROG should not be billed for NRC review fees associated with these potential issues The NRC has a process for addressing potential issues associated with NRC-approved NSSS vendor analytical methods that should be followed, if an issue is identified

Neither the Final Safety Evaluation Report (FSER) for the AP1000 Core Reference Report, nor the APR1400 FSER contained any Limitations and Conditions regarding the effects of EOL conditions nor did it contain legacy issues associated with the NRC approve NSSS vendor analytical methods

Therefore, the PWROG requests that the NRC delete the specific text in Section 4 5, Section 4 6 in its entirety and the associated Limitations and Conditions, Number 3 and 4 in the DSE

Section 1 4 of the Topical Report (TR) identified the seven (7) specific items for which NRC approval was requested The PWROG requests that these 7 items be identified in Section 1.0, "Introduction," and their approval discussed in Section 6.0 "Conclusions," in the DSE

The DSE has been revised in several locations to clarify the use of the terminology "method," and "methodology." Where appropriate, these terms were replaced with "test protocol," technique," and "approach," to reflect the purpose of the TR and to provide consistency with other sections of the DSE that use the appropriate terms

The PWROG requests that the Staff revise the DSE to address these comments and provide a copy of the revised DSE for PWROG review, and that the NRC contact the PWROG with any questions or concerns regarding the PWROG comments

Correspondence related to this transmittal should be addressed to.

Mr. W Anthony Nowinowski, Executive Director
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Sincerely yours,

Ken J Schrader

Ken Schrader
Chief Operating Officer & Chairman
Pressurized Water Reactor Owners Group

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Enclosures (1):

1. Proprietary markups of Draft Safety Evaluation for PWROG report "PWROG-16043-P, Revision 2

cc: PWROG Steering Committee Representatives
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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR TOPICAL REPORT PWROG-16043-P, REVISION 2,

"PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09:

'IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH'

FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS"

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

1.0 INTRODUCTION

By letter dated February 1, 2017 (Reference 1), the Pressurized Water Reactor (PWR) Owners Group (PWROG or the applicant), submitted to the U.S. Nuclear Regulatory Commission (NRC) staff for review licensing topical report (TR) PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs" (Reference 2, henceforth referred to as the TR). Subsequent letters dated March 27, 2018, May 15, 2018, and May 15, 2018 (References 3, 4, and 5, respectively), provided additional information that supplemented the information provided in Reference 2. The TR will be used as the basis for determining fuel assembly characteristics and damping coefficients at End of Life (EOL) conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods~~is an extension of the previously approved methodologies~~ described in WCAP-9401-P-A (Reference 6) and CENPD-178(P), Rev. 1-P (Reference 7), to assess the structural integrity of fuel assemblies under faulted condition loads. ~~The methodologies described in the TR can be used to develop fuel assembly characteristics and damping coefficients for end of life (EOL) conditions that can then be used with the existing testing and analysis methodologies for seismic and loss-of-coolant accident (LOCA) events.~~

Commented [a1]: See Section 1.3 of the TR.

Commented [a2]: This text is not needed due to the revision to the previous sentence above.

2.0 BACKGROUND

Seismic and LOCA events can result in external forces applied to the fuel assemblies (e.g., shaking and/or vibratory forces). Therefore, applicants must evaluate the fuel assembly structural response under these conditions to ensure that regulatory requirements are met with respect to control rod insertability and core coolability. In particular, the spacer grid performance is assessed to determine if plastic deformation is expected to occur, and the fuel assembly vibration behavior is quantified. Most PWR plants currently utilize the NRC approved testing and analysis methodologies described in References 6 and 7 for Westinghouse and CE fuel designs, respectively.

The NRC reviewed and approved References 6 and 7 based on the regulatory guidance provided in Appendix A to Chapter 4.2 of the Standard Review Plan (SRP or Reference 8). One assumption in the SRP Chapter 4.2 Appendix A guidance at the time, which is also in the current revision from 2007, is that beginning of life (BOL) is the time at which the crushing load for the spacer grids would be expected to be at a minimum. This assumption was based on the fact that irradiation tends to cause strengthening in metals and alloys in addition to embrittlement. Other effects that arise due to use in a reactor may include growth, cladding creep, and corrosion. The increase in strength was expected to more than offset the other

Enclosure

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effects associated with irradiated grids. Since applicants typically verify that the maximum load experienced by the spacer grids during LOCA and seismic events will not exceed the crushing load, use of BOL characteristics was considered to be conservative. Operating experience that came to light in the mid-2000s led the NRC staff to question the assumption that the spacer grid structural performance during LOCA and seismic events would not degrade significantly as a result of irradiation. The NRC subsequently issued Information Notice (IN) 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" (Reference 9). This IN lists several factors that can affect the structural strength of the spacer grids, and singles out spacer grid spring relaxation as one that can have a significant effect on the fuel assembly mechanical characteristics and the spacer grid strength. While no specific action or response was required as a result of the IN, the NRC indicated that recipients would be expected to review the information for applicability and consider appropriate action to avoid similar problems.

This TR is the applicant's proposed approach to generically address the issue identified in the IN for licensees that use Westinghouse or CE fuel. The TR will be used as the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods described in References 6 and 7, to assess the structural response of fuel assemblies under faulted condition loads. ~~In essence, this TR describes how to extend the testing and analysis methodologies in References 6 and 7 to determine an appropriate crushing load for spacer grids at EOL. In addition, the TR proposes a methodology that can be used to develop flowing water damping ratios that can then be credited in the LOCA and seismic analyses~~Crediting flowing water damping ratios in a similar manner to the NRC approved still water damping ratios (as described in References 6 and 7), ~~This~~ provides a means for licensees to recover margin lost due to the effect of spacer grid spring relaxation on the fuel assembly mechanical characteristics.

Commented [a3]: See Section 1.3 of the TR.

In summary, the existing NRC approved testing and analysis methodologies will continue to be used, with all previously established limitations and conditions, however ~~but~~ this TR ~~extends the applicability of the relevant aspects of these methodologies to the extent necessary~~provides the basis for determining fuel assembly characteristics and damping coefficients to address potential fuel assembly structural performance issues as a result of irradiation.

3.0 REGULATORY EVALUATION

Title 10, "Energy," of the *U.S. Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," contains requirements for the emergency core cooling system (ECCS) at commercial power plants. In particular, 10 CFR 50.46(b)(4) requires that "[c]alculated changes in core geometry shall be such that the core remains amenable to cooling." Any failure in the structural integrity of the fuel assemblies will typically change the core geometry, and the possibility needs to be evaluated.

The regulation at 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 10, "Reactor design," states that "[t]he reactor core... shall be designed with appropriate margin to assure that specified fuel design limits are not exceeded during...anticipated operational occurrences." Within the context of LOCA and seismic events, this is implicitly addressed by ensuring adequate core coolability.

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The regulation at 10 CFR Part 50, Appendix A, GDC 27, "Combined reactivity control systems capability," states that "[t]he reactivity control systems shall be designed to… reliably [control] reactivity changes…" One of the primary reactivity control systems at current WEC and CE PWR plants is the rapid insertion of control rods to add sufficient negative reactivity to shut down the reactor. Reliable operation of this reactivity control system is conditional on the capability to insert the control rods. Vibrations or structural deformations may impede the control rod movement, and need to be evaluated.

The regulation at 10 CFR Part 50, Appendix A, GDC 35, "Emergency core cooling," restates the requirement to maintain adequate emergency core cooling capability, which can be affected by the core geometry as discussed in 10 CFR 50.46(b)(4) (see above).

The regulation at 10 CFR Part 50, Appendix A, GDC 2, "Design bases for protection against natural phenomena," requires safety-related structures, systems, and components (SSCs), including reactor fuel, to be designed to withstand natural phenomena (such as earthquakes) without a loss of capability to perform safety functions. This GDC also requires consideration of "appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena." For example, a LOCA may be caused by a seismic event, so consideration of the effects from a combination of these two events may be appropriate.

~~Appendix S of 10 CFR Part 50 provides additional guidance for seismic events, and defines the Safe Shutdown Earthquake (SSE), Operating Basis Earthquake (OBE), and safety requirements for relevant SSCs. In general, stress, strain, and/or deformation limits should be defined for each SSC to ensure its functional capabilities during each event indicated by the regulatory requirements (typically OBE, LOCA+SSE, and SSE only, though other combinations may be considered). These requirements are not explicitly addressed by the methodologies submitted for NRC review, however, the overall methodology that PWROG-16043 will supplement is intended to demonstrate that these requirements are met. Therefore, the NRC staff considered the potential impact of PWROG-16042 on how the 10 CFR Part 50 Appendix S requirements would be met.~~

In summary, the NRC Staff used the applicable acceptance criteria for the structural response of fuel assemblies defined in Section 4.2, Appendix A of the SRP otherwise known as NUREG-0800 (Reference 8), in its review of the TR. ~~The acceptance criteria for the structural response of fuel assemblies to externally applied forces, in order to satisfy the above criteria, are defined in Section 4.2, Appendix A of the SRP, otherwise known as NUREG-0800 (Reference 8). In general, t~~The primary criteria are related to ensuring that core coolability and control rod insertability are maintained.

~~This TR is an application of an evaluation model to perform licensing analyses for an accident that the evaluation model has not previously been approved. As such, additional guidance for the evaluation may be found in SRP Chapter 15.0.2, "Review of Transient and Accident Analysis Methods" (Reference 10). This chapter includes provisions for the review of submittals related to evaluation models.~~

~~In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 along with the applicable acceptance criteria in SRP Chapters 4.2 Appendix A in conducting its review of the TR. Since the TR is effectively a supplement to existing methodologies, the scope of the NRC staff review was limited to the elements of the TR that represented a novel approach relative to~~

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~~the existing methodologies, and to verify the applicability of the existing methodologies when conducting tests and evaluations as described in the TR.~~

4.0 TECHNICAL EVALUATION

The intent of the TR is to develop the basis for determining fuel assembly characteristics and damping coefficients at End of Life (EOL) conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods~~avoid extensive modification of previously approved analysis methodologies documented~~ in References 6 and 7 by focusing solely on the specific parameters that would be impacted by the EOL issues identified in IN 2012-09 (Reference 9). As such, the TR narrowly focuses on three primary parameters:

1. The allowable grid impact strength [ ]
1. The fuel assembly modal frequencies [ ] and
1. ~~The~~Crediting fuel assembly flowing water damping ratio, [ ]

As a result, some of the areas from SRP Chapter 15.0.2 are not applicable. In particular, the analysis methods~~ologies~~ described in References 6 and 7 are not being modified, only the empirical determination of key input parameters. Therefore, the accident scenario description, the phenomena identification and ranking, and code assessment from the previously approved methods~~ologies~~ remain valid. The NRC staff review of the TR focused on two of the specific areas described in SRP Chapter 15.0.2, as described below:

1. Evaluation methodology – the proposed testing and data analysis approach~~methodologies~~, including any potential limitations to their applicability.

2. Uncertainty analysis – the applicant's evaluation and propagation of uncertainties in the analysis of test data to obtain recommended values for the key parameters.

In addition, the NRC staff considered whether the applicant provided adequate quality assurance (QA) and documentation support for the proposed approach for addressing the EOL effects on spacer grids~~methodologies~~. This aspect is not explicitly discussed in detail for this safety evaluation (SE) because the documentation of the proposed approach~~methodologies~~ is~~are~~ captured by the documents reviewed by the NRC during an audit dated October 17, 2017 (Reference 11) and that were found to have been appropriately summarized or otherwise characterized in the TR. The testing was performed under the auspices of the same QA program for~~used to perform the~~ testing ~~for the~~ previously performed~~approved methodologies~~ to determine the key parameters for BOL grids and still water damping, which is acceptable. As

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such, the NRC staff acceptance of the adequacy of the applicant's test protocol~~evaluation methodologies~~ and uncertainty analyses implicitly includes acceptance of the applicant documentation associated with that area.

4.1 EOL Grid Simulation

~~All of the proposed methodologies in the~~This TR discusses the test protocol used for the~~are based on a specific~~ characterization of the impact of irradiation on the spacer grids. SRP Chapter 4.2 Appendix A (Reference 8) cites several possible irradiation-related effects relevant to spacer grids, and concludes that the combined impact would not be expected to lead to a more conservative result. This logic rests mainly on the fact that the significant increase in yield strength for the spacer grid material will more than offset the relatively minor effects from the remaining effects. As described in IN 2012-09 (Reference 9), operating experience has shown that spacer grid spring relaxation can have a significant adverse effect on spacer grid strength and fuel assembly mechanical characteristics. [

] Other than grid spring relaxation, the basic assessment in SRP Chapter 4.2 Appendix A that irradiation-related effects are bounded by the increase in the yield strength of the spacer grid material continues to be applicable. [

]

[

] As discussed in the previous paragraph, the NRC staff found that the focus on the grid spring relaxation phenomenon as the key driver for the non-conservative behavior identified in spacer grids at EOL relative to BOL is appropriate. However, the material and geometry impacts of the thermal relaxation process must be reasonably similar to the irradiation-induced impacts that are being simulated.

[

]Therefore, the NRC staff requested additional information from the applicant regarding the thermal relaxation procedure used to produce the simulated EOL grids. [

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Commented [a7]: See Section 1.2 of the TR

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] The applicant's response also confirmed that the material structural characteristics of the simulated EOL grids are the same, or slightly conservative, relative to the BOL grids.

[

] There are some situations where a spacer grid is exposed to a strongly non-uniform neutron flux, such as fuel assembly loading locations at or near the core periphery. The NRC staff asked the applicant to address the potential impact on the grid failure mechanism due to non-random gradients in gap size that may be correlated with steep neutron flux gradients. [

]

Finally, ~~Section~~Chapter 2.1 of the TR described how the target average gap size was determined for a given spacer grid. [

]

Inadequate information was given in the TR to define the area of applicability for extrapolation of a given set of PIE data to the general population of EOL grid spacers of the same design, so the NRC staff requested that the applicant characterize how PIE data sets are generally defined in order to achieve their intended purpose.

The applicant responded in Reference 4 with an explanation of the statistical ~~technique~~methodology underlying their determination of a target gap size for the simulated EOL grids. [

] this is a reasonably conservative approach to ensure that the average gap sizes for the simulated EOL grids will bound the average gap sizes for irradiated grids.

[

] The NRC staff agrees, however, the applicant did not describe how the rod burnups associated with

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the PIE measurements would be used to define the area of applicability for fuel assemblies qualified using this approach~~under this methodology~~. In a separate RAI response (RAI-2, documented in Reference 4), the applicant provided information that shows that the variation in gap sizes for varying burnups near EOL can be expected to be minor relative to the inherent randomness in gap sizes within a grid. In addition, the NRC staff noted that the protocol~~methodology~~ described in Reference 7 for testing of CE design fuel assemblies includes modeling for both BOL and EOL grids. [

] Consistent with this assessment, the results from the testing discussed in Sections 4.2 and 4.3 of this SE show [

] Therefore, any variations in burnup for the fuel assemblies used to obtain PIE measurements relative to the overall population of fuel assemblies being qualified using~~under~~ this approach~~methodology~~ would not result in a significant difference in average gap size, certainly, much less than the inherent conservatism in the margin between the average measured gap sizes and the target gap size for the simulated EOL grids.

[

] As a result, the NRC staff found the proposed approach to generate simulated EOL grids for use in testing in lieu of irradiated grids to be acceptable.

4.2 Spacer Grid Impact Strength

Sections~~Chapters~~ 2.2 and 2.3 of the subject TR discuss the application of the approved testing and data analysis protocol~~methodologies~~ from References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. [

]The NRC staff understanding of the approval request from the applicant is that this additional~~change in~~ criterion was provided~~adopted~~ merely for demonstration purposes, and was not ~~being~~ submitted as a change~~an update~~ to how the grid impact strength is determined in ~~the~~ Reference 6 ~~methodology~~. In response to a RAI from the NRC staff (Reference 3), the applicant confirmed that this was the case. Therefore, this application was judged to be acceptable solely for the purpose of providing a more consistent basis for comparing ~~the change in~~ P(crit) for Westinghouse and CE fuel designs.

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The simulated EOL grids contain [ ] The NRC staff verified by inspection of the applicant's test documentation that the failure mechanism for the simulated EOL grids was the same as that for the BOL grids. Therefore, [ ] As discussed in Section 4.2 of this SE, [ ]

The NRC staff verified that the previously approved testing and data analysis protocols in~~methodologies from~~ References 6 and 7 were appropriately applied to the simulated EOL grids. In addition, the NRC staff found reasonable assurance exists that the aforementioned test protocols~~methodologies~~ remain applicable to the geometry of the simulated EOL grids. Therefore, the NRC staff found the approach for determining~~methodologies to determine~~ P(crit) to be acceptable for use in analysis of the simulated EOL grids.

4.3 Fuel Assembly Mechanical Characteristics

Section~~Chapter~~ 3 of the TR discusses the application of the approved testing and data analysis protocols in~~methodologies from~~ References 6 and 7 to determine the allowable grid impact strength for the simulated EOL grids. The TR states that "[t]he same test protocol has been previously applied to current Westinghouse and CE PWR fuel designs for BOL conditions," and that "[t]he test protocols are described in NRC-approved TRs..." with a citation to References 6 and 7. Therefore, the TR clearly characterizes the testing procedure for the simulated EOL grids to be identical to the previously approved testing procedure described in References 6 and 7, with the exception that the grids are simulated EOL grids as discussed in Section 4.1 of this SE.

The testing protocols~~methodologies~~ described in References 6 and 7 are primarily tests conducted on the structural members of the fuel assembly and the spacer grids, with no tests directly impacting the fuel rods. At BOL, the grid springs exert a frictional force on the fuel rods, so the spacer grids and fuel rods are mechanically coupled to some extent. During the fuel assembly vibration tests, the fuel rods contribute to the fuel assembly mechanical performance by virtue of this mechanical coupling. [ ]

[ ]

4.4 Procedure to Determine Flowing Water Damping Ratios

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Section~~Chapter~~ 4 of the TR describes the test protocol for determining the~~a methodology to determine~~ fuel assembly flowing water damping ratios and apply them in lieu of previously approved still water damping ratios to characterize the fuel assembly mechanical behavior during seismic and LOCA events. Since the damping ratio due to flowing water is expected to be higher than that for still water, this approach could help recapture margin lost due to the impact of grid spacer relaxation on the fuel assembly stiffness. [

]

Section~~Chapters~~ 4.1 through 4.3 describe the test apparatus and data collection performed to support an empirical determination of the flowing water damping ratios. [

] However, the hydraulic characteristics for the fuel assembly are well characterized based on prior testing. [

] Since the loss coefficients for the fuel assembly designs have been approved by the NRC for use in other analyses and would not be expected to vary significantly as a result of the use of simulated EOL grids, this approach for determining flow velocities through the fuel assembly is acceptable.

The existing test protocols~~analysis methodologies~~, most notably the Reference 7 protocol~~methodology~~ for CE fuel, [

] Testing performed on similar fuel assembly designs using a range of different approaches, as documented in References 14 and 15, yield consistent results. [

] This shows that the proposed approach discussed in this TR~~methodology~~ yields results consistent with that currently approved in Reference 13~~methodologies~~.

The flowing water damping ratio correlation was developed based [

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] Therefore, there will be no inconsistency in the application of damping ratios for fuel assemblies at different burnup conditions.

Based on the data collected from the tests, a damping ratio was determined for each test based on classical vibration theory. [

] Chapter 4.5 of the TR presents results from the tests. One of the most important conclusions that can be observed directly from the test results is that [

]

Since the use of lower damping ratios in developing the correlation is conservative, this was an acceptable choice to make.

SectionChapter 4.6 of the TR discusses the data analysis approach used to determine bounding correlations for each fuel assembly design. This approach can be summarized thus:

[

] The overall approach appears to capture the relevant dependencies, however, there is no propagation of the uncertainties due to scatter in data through the steps noted above. [

]

The applicant responded in Reference 5 with information indicating that the fitting approach used to determine the bounding curve was fundamentally a best estimate approach to derive the 600 °F curve based on the selected data set. [

]

[

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]

Finally, Chapter 4.7 proposes use of a flowing water damping ratio correlation based on the [ ] fuel assembly design as a generically bounding correlation that may be used with any fuel assembly design without further justification. The ~~procedure~~methodology discussed above may be used to develop fuel assembly design specific correlations, but the [ ] correlation is proposed for use as a bounding curve for all Westinghouse and CE fuel designs. The justification provided is that the [ ] fuel assembly design proposed for the [ ] reference plant contains a number of significant design differences, but test results show that the flowing water damping ratio is very similar to the [ ] fuel. The CE fuel design tested had [

] This behavior is bounded by the [ ] correlation, so this is acceptable. However, [

] Therefore, the similarity in results is not surprising.

In order to establish that the proposed correlation can be used as a generic bounding curve, its applicability must be limited to spacer grids with very similar geometry characteristics. This is accomplished via a condition to the TR. Information submitted in References 14 and 15 provide information for other PWR fuel assembly designs that suggests that, in fact, the [

] As long as the geometry characteristics of the spacer grids associated with a different fuel assembly do not differ significantly from the RFA/RFA-2 spacer grid, the NRC staff finds that reasonable assurance exists that other fuel assembly designs will have flowing water damping ratios near or above the proposed bounding curve. The proposed application includes use of a minimum value for the analysis duration rather than a more realistic average value, which incorporates some additional conservatism that offsets the potential for slightly lower flowing water damping ratios for some fuel assembly designs relative to the proposed bounding curve.

Based on the information provided in the TR, as supplemented by responses to requests for additional information from the NRC staff, the testing protocol and data analysis ~~methodologies~~ described to determine appropriate flowing water damping ratios were determined to be appropriate for their intended purpose. In addition, [

] This latter condition was captured in Section 5.0

4.5 Analytical Application of the Flowing Water Damping Ratios

~~Sections~~Chapters 4.8 and 4.9 of the TR describe when and how the flowing water damping ratios can be utilized in seismic and LOCA analyses, respectively. The primary parameter used to establish the appropriate value for the flowing water damping ratio is the fluid velocity through the fuel assembly. For a given plant, this parameter is directly correlated with the core flow.

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Therefore, the discussion in the TR primarily focuses on the characterization of a bounding core flow for any given time of interest during the event being analyzed. Once an appropriate value is determined, then plant-specific information can be used to establish an appropriate flow velocity to use with the flowing water damping ratio correlation. [

] In general, since lower flow velocities result in lower flowing water damping ratios, any factor that may lead to a reduction in the core flow rate will provide more conservative results. For a given analysis, [

]

For the seismic analysis, two key assumptions are made to minimize the total core flow. First, [

]

Secondly, [

] At this time, the flowing water damping ratio will be at a minimum, and lower than the average flowing water damping ratio for the interval. Since these assumptions both act to minimize the flowing water damping ratio, they are conservative.

For the LOCA analysis, the core flow rates are to be obtained directly from the LOCA analyses as long as axial flow is maintained. [

] As a result, the NRC staff finds that the LOCA analysis conditions are an acceptable source for a bounding core flow rate for the purpose of determining flowing water damping ratios.

A second limitation of the flowing water damping ratios is that the data used as a basis for the correlation were based on single phase liquid flow through a fuel assembly. The conditions under which the flowing water damping ratios are expected to be credited—seismic events and the first ~1 second of a LOCA event—are not expected to involve two phase flow in the core. However, the TR does not explicitly limit the use of flowing water damping ratios to single phase flow conditions, so a limitation was included in Section 5.0 to ensure that, if credit for flowing

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water damping~~this methodology~~ is applied to conditions that deviate from expectations, the correlation will not be used outside the bounds of its applicability.

~~The NRC approval of Reference 13 included review of information demonstrating that the Westinghouse models were capable of capturing the dynamic behavior of fuel assemblies for pluck response inside a flow loop, for the vibration range of interest. Since the flowing water damping ratios are very similar for the RFA/RFA-2 curves being proposed for use as a bounding curve for all fuel assembly designs and the Reference 13 fuel design contained a similar spacer grid design, this finding is applicable to the subject LTR as well. However, without further validation, the dynamic models cannot be assumed to maintain reasonable accuracy for damping ratios that go significantly beyond the current area of applicability. Therefore, any use of damping ratios significantly higher than the proposed bounding curve must be supported by validation against test data that demonstrates that the analytical models remain valid for the higher damping regime. A limitation was included in Section 5.0 to ensure that this potential limitation of the analytical models is addressed, if necessary.~~

The guidance provided in the TR to credit flowing water damping in seismic and LOCA analysis was reviewed by the NRC staff and determined to produce acceptably conservative results for the expected analysis conditions. Therefore, the NRC staff finds the proposed application of flowing water damping credit for evaluation of fuel assembly mechanical behavior during seismic and LOCA events to be acceptable.

~~4.8 Known Legacy Issues~~

~~There are a number of potential issues with the previously approved methodologies described in References 6 and 7. They include:~~

[

]

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~~These issues may have been addressed for legacy fuel assembly designs based on expected fuel assembly grid behavior and testing. However, the current approved methodologies do not provide a generic approach to do so. Therefore, the assumptions inherent in the technical justification for these issues need to be evaluated on a case-by-case basis for new fuel assembly designs, which may depend on consideration of all attributes of the proposed revisions to the plant licensing basis. The new proposed approach to credit flowing water damping ratios represents a more realistic approach. As such, there is a reduction in conservatism for this approach relative to the previously approved approach to credit still water damping. Therefore, the overall justification for the above issues must be re-evaluated to ensure that the overall analysis remains conservative.~~

~~As discussed above, the NRC staff identified some technical issues that are not explicitly addressed by the currently approved methodology. They may have been addressed for current fuel assembly designs, however, the use of a more realistic flowing water damping ratio represents a reduction in conservatism for the damping ratio approach relative to the previously approved approach. Therefore, the NRC staff is imposing limitations and conditions to ensure that the overall conservatism of the analysis is acceptable.~~

5.0 LIMITATIONS AND CONDITIONS

Some limitations and conditions are necessary to ensure that the approach to address NRC IN 2012-09~~methodology~~ discussed in the TR is limited to the applications for which it is valid. These limitations and conditions are listed below.

1. [

]

6.0 CONCLUSIONS

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~~In the TR, the applicant presented new models and methods to extend the applicability of existing methodologies to evaluate spacer grid and fuel assembly mechanical behavior during seismic and LOCA events.~~ The applicant submitted a TR that will be used as the basis for determining fuel assembly characteristics and damping coefficients at EOL conditions for input into plant specific seismic and LOCA analyses that will be performed in accordance with the current NRC approved methods described in References 6 and 7, to assess the structural integrity of fuel assemblies under faulted condition loads. The following conclusions are provided here in summary as they apply to licensees who may want to adopt the ~~methodologies described in~~ the TR ~~with existing methodologies in References 6 and 7~~ to address the effect of irradiation on the mechanical properties of fuel assemblies.

Since the TR is not proposing any change to the previously approved testing and analysis methods~~ologies~~ for seismic and LOCA events, the NRC staff performed a graded review of the TR~~methodologies~~ that took into consideration the fact that most aspects of this TR~~the testing and analysis~~ have already been addressed as part of prior NRC reviews. The applicant requested approval of seven specific items identified in Section 1.4 of the TR ~~for three distinct enhancements to their existing methods: (1) use of simulated EOL spacer grids to assess spacer grid crush strength at EOL; (2) use of simulated EOL spacer grids to assess fuel assembly mechanical characteristics, such as stiffness, at EOL; and (3) use of a new methodology to develop flowing water damping ratios that can be used in lieu of the currently approved still water damping ratios.~~

The NRC staff examined the proposed approach to produce simulated EOL spacer grids ~~and use them with previously approved methodologies,~~ and determined that the simulated EOL spacer grids would adequately capture the non-conservative impacts due to irradiation. The staff also determined that the [

] The NRC staff's findings were based primarily on the specific material type (zirconium alloy) and general grid design covered by the information presented in the TR. [

]

The use of flowing water damping ratios is not an entirely new approach to develop more realistic parameters that help mitigate the impact of vibratory loads, because it is similar to what was approved by~~a methodology submitted as part of~~ the NRC for~~approval of~~ the AP1000 ~~reference plant design~~ (Reference 1~~3~~4). However, this is the first time that it is being applied more generically to Westinghouse and CE fuel. In particular, the applicant is proposing the use of a bounding curve that is applicable to all spacer grids used in Westinghouse and CE fuel, along with a general approach~~methodology~~ that can be used to generate fuel design specific curves. The staff reviewed the information submitted in the TR along with responses to requests for additional information, and determined that the approach~~methodology~~ was appropriate for both purposes. Additionally, the guidance provided for utilization of flowing water damping ratios in seismic and LOCA analyses was found to be appropriate for their intended use, with the ~~l~~Limitations and Condition 2 above~~that: (1) the flowing water damping ratios are only valid for single phase liquid flow; and (2) the dynamic models used to predict the fuel assembly response under vibratory and damping loads must be verified to remain~~

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~~reasonably accurate for higher damping regimes by validation against test data, prior to use for safety analysis purposes.~~

~~The NRC staff also acknowledged some legacy issues with lack of clear guidance to address certain aspects of current NRC regulations. Since approval of use of specific fuel assembly designs at specific plants may have depended on consideration of fuel design specific characteristics that would disposition or offset the legacy issues, the NRC staff finds that any reduction in analytical conservatism should not be made without addressing these legacy issues, as discussed in Section 4.5. The use of flowing water damping ratios represents one such reduction in analytical conservatism, therefore, a condition for use of the new damping ratios is that the legacy issues need to be addressed.~~

Commented [a14]: Delete this text, please see the text in letter OG-19-13.

In summary, the NRC staff finds that the information provided in the TR and responses to NRC staff RAIs adequately demonstrates that the proposed approach~~methodologies~~ to address EOL effects on spacer grids and to recover margin through credit for flowing water damping are acceptable for use with existing methods~~ologies~~ that the NRC has previously found to be acceptable for analysis of fuel assembly structural behavior during seismic and LOCA events. The NRC staff approval of this TR~~these methodology~~ extends to all Westinghouse and CE fuel designs, contingent on adherence to the limitations and conditions set forth in Section 5.0.

7.0 REFERENCES

1. PWROG letter OG-17-12, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Submittal of PWROG-16043-P, Revision 2, 'PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs, PA-ASC-1169R2," February 1, 2017 (ADAMS Accession No. ML17039B050)

2. PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 2017 (ADAMS Package Accession No. ML17039B061)

3. PWROG letter OG-18-62, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request for Additional Information, RAIs 4 and 5 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," March 27, 2018 (ADAMS Accession No. ML18100A053)

4. PWROG letter OG-18-104, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request for Additional Information, RAIs 1, 2, and 3 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs,' PA-ASC-1169," May 15, 2018 (ADAMS Accession No. ML18143B462)

5. PWROG letter OG-18-105, Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, to USNRC document control desk, re: "Transmittal of the Response to Request

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- 17 -

for Additional Information, RAI 6 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," PA-ASC-1169," May 15, 2018 (ADAMS Accession No. ML18144A760)

6. WCAP-9401-P-A, Revision 0, "Verification Testing and Analysis of the 17x17 Optimized Fuel Assembly," September 1981 (ADAMS Accession No. ML090280466 (Non-Publicly Available))

7. CENPD-178(P), Revision 1-P, "Structural Analysis of Fuel Assemblies for Seismic & LOCA Loading," August 1981 (ADAMS Accession No. ML14122A086 (Non-Publicly Available))

8. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 4.2, Revision 3, "Fuel System Design," March 2007 (ADAMS Accession No. ML070740002)

9. NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," dated June 28, 2012 (ADAMS Accession No. ML113470490)

10. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 15.0.2, Revision 0, "Review of Transient and Accident Analysis Methods," March 2007 (ADAMS Accession No. ML070820123)

11. NRC letter from Brian Benney, Senior Project Manager, Licensing Processes Branch, Division of Policy and Rulemaking, USNRC, to Jack Stringfellow, Chief Operating Officer and Chairman, PWROG, re: "Summary Report for the October 17, 2017, Audit in Support of the Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: 'Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength' for Westinghouse and CE PWR Fuel Designs," January 8, 2018 (ADAMS Accession No. ML17326A003)

12. Framatome ANP, Inc. letter NRC:03:051, James F. Malley, Director, Regulatory Affairs Framatome ANP, Inc., to USNRC document control desk, re: "Closure of Interim Report 02-002, 'Spacer Grid Crush Strength – Effects of Irradiation,'" August 8, 2003 (ADAMS Accession No. ML032240425)

13. WCAP-17524-P/NP-A, Revision 1, "AP1000 Core Reference Report," May 2015 (ADAMS Accession No. ML15160A175)

14. Westinghouse letter LTR-NRC-13-26, James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, to USNRC document control desk, re: "Supplemental Information on End-of-Life Seismic/LOCA calculations for the AP1000 Pressurized Water Reactor (Proprietary/Non-Proprietary)," April 30, 2013 (ADAMS Accession No. ML13128A017)

15. Framatome Inc. report ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," April 2018 (ADAMS Package Accession No. ML18144A816)

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- 18 -

Principal Contributor: Scott Krepel, NRR/DSS/SNPB

Date: August 22, 2018

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PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-16043-P, Revision 2
Project Number 99902037

May 15, 2018

OG-18-105

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information, RAI 6 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169

References:

1. Letter OG-17-12, Submittal of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," PA-ASC-1169R2, dated February 1, 2017

2. NRC Letter of Acceptance for Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," dated June 20, 2017

3. Email from the NRC (Benney) to the PWROG (Holderbaum), Request for Additional Information, RAIs 1-6, RE: PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," dated January 31, 2018

4. Letter OG-18-62, Transmittal of the Response to Request for Additional Information, RAIs 4 and 5 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169, dated March 27, 2018

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Page 2 of 3

5 Letter OG-18-104, Transmittal of the Response to Request for Additional Information, RAIs 1, 2 and 3 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169, dated May 15, 2018

On February 1, 2017, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of PWROG-16043-P, Revision 2 for referencing in regulatory actions (Reference 1). The NRC Staff has determined that additional information is needed to complete the review per letter dated January 31, 2018 (Reference 3).

Enclosure 1 to this letter provides a response to NRC RAI 6 (Reference 3) associated with PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09 "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs.

Responses to NRC RAIs 4 and 5 were transmitted to the NRC via Reference 4 on March 27, 2018. Reference 5 transmitted responses to NRC RAIs 1, 2 and 3 to the NRC on May 15, 2018.

Also enclosed is the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-18-4739, 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 item listed above or the supporting Westinghouse Affidavit should reference CAW-18-4739 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

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, COO & Chairman
PWR Owners Group

JKS:am

cc: PWROG Analysis Committee (Participants of PA-ASC-1169)
PWROG PMO
PWROG Steering and Management Committee
J. Andrachek, Westinghouse
K. Lasswell, Westinghouse
J. Sinegar, Westinghouse
B. Benney, US NRC

Enclosure 1: PE-18-25-P/NP, Attachment 1, "Response to PWROG Topical Report PWROG-16043-P RAI 6" (PA-ASC-1169)

Enclosure 2: Affidavit for Withholding, CAW-18-4739 (Non-Proprietary) with accompanying Affidavit, Proprietary Information Notice and Copyright Notice

Westinghouse Non-Proprietary Class 3

Westinghouse

From: Roger Yong Lu
Phone: (803) 647-3426
e-mail: lur@westinghouse.com

Memo: PE-18-25-NP Rev. 1
Date: May 9, 2018

Subject: **Response to PWROG Topical Report PWROG-16043-P RAI 6**

To: James P. Molkenthin Jill G. Sinegar James D. Andrachek
cc: PWROG

Attached is the response to RAI 6 related to the PWROG Topical Report PWROG-16043-P. Proprietary information is shown in brackets. Questions or comments should be directed to the undersigned.

Author:	Roger Y. Lu* PWR Fuel Technology
Verifier:	Jane X. Jiang*† Thermal-Hydraulic and Seismic Engineering
Verifier:	Jiwei Wang*† PWR Fuel Technology
Approver:	Kevin T. Lasswell*, Manager Thermal-Hydraulic and Seismic Engineering

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

† Three Pass Verification (3PV) was used to verify this document as demonstrated in the Design Document Verification Checklist which is attached to this document in PRIME.

A new methodology is being proposed for Westinghouse and CE fuel to credit flowing water damping in mitigation of the degradation in fuel mechanic behavior due to EOL effects on the spacer grids. This methodology is proposed as an option for use in lieu of the still water damping credited in the previously approved methodologies. In order to fully understand how the proposed methodology is intended to conservatively capture the impact of flowing water on fuel assembly vibrations, the NRC staff requests the following information:

RAI Item 6

Section 4.6 of the LTR [

] a,c

Response to 6a

The fuel assembly damping ratio is the measurement of energy dissipation in a mechanical system. To account for the energy dissipation during vibration, the averaged or best estimated damping ratio value is more appropriate to a full core fuel assembly analysis from a physical standpoint. This is different from other local bounding analyses, such as a Departure from Nucleate Boiling (DNB) correlation.

Fuel Assembly (for the AP1000® Plant) Flowing Water Damping Background

[ ] a,c

[

] a,c

a, c

Figure 1: Damping vs. Velocity Curve that was used for [ ] a, c Model (Reference 1)

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May 9, 2018
Page 3 of 7

Flowing Water Damping Curve for PWROG-16043-P

[

] a,c

[ ] a, c

Figure 2. Damping vs Velocity Curve for [ ] a,c

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Response to 6b

The fuel assembly damping force in flowing water is the summation of the fuel structural damping in air, viscous damping in still water and the hydraulic damping in flowing water as shown in Equation (1) The flowing water damping coefficient measured and used in PWROG-16043-P is also the summation of these three components

$$F_d = c_s\dot{x} + c_v\dot{x} + c_h\dot{x} \qquad (1)$$

c_s – The structural damping coefficient in air, due to material and friction damping.
c_v – The viscous damping coefficient in still water
c_h – The hydraulic damping coefficient in flowing water, which primarily increases with the axial flow velocity

All three damping coefficients in Equation (1) are neither constant nor linear. All tests that were performed by other fuel vendors concluded that the water temperature has a small effect on fuel assembly damping. Babcock & Wilcox's paper (Reference 2) concluded that damping is minimally affected by temperature ranges from 68°F to 600°F. The Mitsubishi Heavy Industries' topical report (Reference 3) concluded "that the temperature effect of AFD (Axial Flow Damping) appears to be very small up to the reactor operating condition from the maximum test temperature." The flowing water damping tests performed by Westinghouse are consistent with this conclusion

Test data trend curve fitting

[

] a, c

1) [

] a, c

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[ ]a, c

Figure 3: Damping vs Density at [ ]a, c (Figure 4-14 of PWROG-16043-P)

2) [ ]a, c

Table 1: The average damping ratios at different temperatures at [ ]a, c

[ ]a, c

3) [ ]a, c

A discussion of the conservatism in the 600°F damping curve

[ ]a, c

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[

] a, c

[ ] a, c

Figure 4: Damping Ratio vs. Coastdown Time
for a Typical Westinghouse 3-Loop Unit (Figure 4-21 of PWROG-16043-P)

Summary and Conclusions

[

] a, c

References

1. [

]a, c

2. F. E Stokes and R. A. King, "PWR Fuel Assembly Dynamic Characteristics," International Conference on Vibration in Nuclear Power Plants, Keswick, United Kingdom, May 9-12, 1978 (BNES), Page 31.

3. MUAP-13020-NP (R0), "Axial Flow Damping Test of the Full Scale US-APWR Fuel Assembly," August 2013, Page 3-2.

4. WCAP-9401-P-A, "Verification Testing and Analyses of the 17 x 17 Optimized Fuel Assembly," August 1981.

PE-18-25-NP Revision 1 Proprietary Class 3

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Approval Information
Author Approval Lu Roger May-10-2018 10:41:07
Verifier Approval Wang Jiwei May-10-2018 10:50:51
Verifier Approval Jiang Jane May-10-2018 11:56:12
Manager Approval Lasswell Kevin T May-10-2018 16:07:37

Files approved on May-10-2018

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Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-16043-P, Revision 2
Project Number 99902037

May 15, 2018

OG-18-104

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information, RAIs 1, 2 and 3 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169

References:

1. Letter OG-17-12, Submittal of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," PA-ASC-1169R2, dated February 1, 2017

2. NRC Letter of Acceptance for Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," dated June 20, 2017

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May 15, 2018
Page 2 of 3

5. Letter OG-18-105, Transmittal of the Response to Request for Additional Information, RAI 6 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09· "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169, dated May 15, 2018

On February 1, 2017, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of PWROG-16043-P, Revision 2 for referencing in regulatory actions (Reference 1). The NRC Staff has determined that additional information is needed to complete the review per letter dated January 31, 2018 (Reference 3).

Enclosure 1 to this letter provides a response to NRC RAIs 1, 2 and 3 (Reference 3) associated with PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs.

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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 item listed above or the supporting Westinghouse Affidavit should reference CAW-18-4738 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.

U.S. Nuclear Regulatory Commission
OG-18-104

May 15, 2018
Page 3 of 3

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, COO & Chairman
PWR Owners Group

JKS:am

cc: PWROG Analysis Committee (Participants of PA-ASC-1169)
PWROG PMO
PWROG Steering and Management Committee
J. Andrachek, Westinghouse
K. Lasswell, Westinghouse
J. Sinegar, Westinghouse
B. Benney, US NRC

Enclosure 1. PE-18-34-P/NP, Attachment 1, "RAIs 1, 2 and 3 Responses for PWROG-16043 Revision 2" (PA-ASC-1169)

Enclosure 2: Affidavit for Withholding, CAW-18-4738 (Non-Proprietary) with accompanying Affidavit, Proprietary Information Notice and Copyright Notice

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Westinghouse Non-Proprietary Class 3

Westinghouse

From: Jane Xiaoyan Jiang
Phone: (803) 647-3735
e-mail: jiangx@westinghouse.com

Memo: PE-18-34-NP
Date: May 8, 2018

Subject: **Response to PWROG Topical Report PWROG-16043-P RAIs 1, 2 and 3**

To: Jill G. Sinegar James D. Andrachek James P. Molkenthin
cc: PWROG

Attached are the responses to RAIs 1, 2, and 3 related to the PWROG Topical Report PWROG-16043-P. Proprietary information is shown in brackets. Questions or comments should be directed to the undersigned.

Author: Jane X. Jiang *
Thermal-Hydraulic and Seismic Engineering

Verifier: Roger Y. Lu *†
PWR Fuel Technology

Approver Kevin T. Lasswell, Manager*
Thermal-Hydraulic and Seismic Engineering

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

† Three Pass Verification (3PV) was used to verify this document, as demonstrated in the Design Document Verification Checklist which is attached to this document in PRIME.

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PE-18-34-NP,
Attachment 1
May 8, 2018
Page 1 of 11

RAI 1

The most significant aspect of the proposed methodology to address EOL effects on the spacer grids is the use of simulated EOL grids, which are grids that have been [[ ]]$^{a, c}$ to simulate the most important non-conservative EOL effect due to irradiation, grid spring relaxation. In order for the simulated EOL grids to accurately capture the limiting behavior of irradiated grids, the structural characteristics of the simulated EOL grids must be similar to, or more conservative than, the irradiated grids In order to verify this, the NRC staff requests the following information

The [[ ]]$^{a, c}$ protocol is not detailed in the LTR. In Section 2.1, the LTR states that the "process for compiling PIE data and specifying target cell size is consistent with that was used for the AP1000 EOL issues that was previously approved by the NRC " However, the exact [[ ]]$^{a, c}$ protocol is unclear Please provide the specifications for the [[ ]]$^{a, c}$ process, including [[ ]]$^{a, c}$

Response to RAI 1

The [ ]$^{a, c}$ of the simulated End of Life (EOL) grids for the AP1000® plant and the simulated EOL grids discussed in PWROG-16043-P was performed in accordance with a Westinghouse thermal cell sizing procedure. The procedure is used to thermally size grid cells to simulate EOL grid conditions for fuel assembly hydraulic loop tests

[ ]$^{a, c}$. The process is shown in Figure 1. [ ]$^{a, c}$

The mechanical structure characteristics of simulated EOL grids is similar to, or more conservative than, the irradiated grids. [ ]$^{a, c}$ Therefore, the grid material characteristics of Young's modulus and Poisson's ratio are not impacted by the grid [ ]$^{a, c}$ process The Young's modulus is one of main parameters which determine the grid impact stiffness.

[ ] a, c

Figure 1. [ ] a, c

Since the [ ] a, c This may result in a slight reduction in the grid impact strength, which is conservative.

The [ ] a, c target cell sizes or gap sizes are varied depending on the Post Irradiation Examination (PIE) data and types of grid designs. [ ] a, c

1. [.] a, c
2. [ ] a, c

Overall, the [ ] a, c for cell sizes will have no impact or a minimally conservative impact on the grid strap material mechanical properties

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RAI 2

Fuel assemblies that are loaded in certain areas of the core may experience steep radial neutron flux gradients. As such, the EOL effects due to irradiation of the spacer grids may not be sufficiently uniform to result in spacer grid behavior consistent with simulated EOL grids using the [[ ]]$^{a, c}$ method. Please characterize the expected variation due to radial neutron flux gradients in typical PWR cores, and discuss how this may impact the spacer grid structural behavior (e g., if gaps exist at one corner of a fuel assembly but not at the opposite corner, explain what the effect on the failure mechanism might be).

Response to RAI 2

a. **"expected variation due to radial neutron flux gradients in typical PWR cores"**

The fuel rods in a fuel assembly may experience steep radial neutron flux gradients in some core locations during some cycles. However, the grid gap size formation (due to grid spring relaxation, rod diameter creep and grid growth) is a long-term and slow process which occurs over the entire irradiated life of a fuel assembly

The typical irradiated lifetime of a fuel assembly is at least 4 years during which it will be rotated to different locations in the core and experience different flux gradients and orientations. Therefore, a radial neutron flux gradient effect on the grid cell size at the fuel assembly EOL condition is not expected to occur.

To confirm that the neuron flux gradient effect does not occur, two sets of PIE data, fuel rod burnup vs cell gap size, were reviewed Fuel rod burnup at EOL is the accumulated effect of neutron flux. [

].$^{a, c}$

The first example is a [ ]$^{a, c}$ for which the measured gap results and corresponding fuel rod burnups are given in Figure 2 A sample of ten fuel rods in different locations in the fuel assembly with different fuel rod burnups is shown in Figure 2 [

]$^{a, c}$

Figure 2 also shows that [

.]$^{a, c}$

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a, c

Figure 2. Measured [ ]a, c

The second example is for a [ ]a, c The measured gap results and corresponding fuel rod burnups are given in Figure 3. A sample of ten fuel rods in different locations in the fuel assembly with different fuel rod burnups is shown in Figure 3. [

]a, c

Figure 3 also shows that [

]a, c

a, c

Figure 3. Measured [ ]a, c

*** This record was final approved on 5/14/2018 10 48 53 AM (This statement was added by the PRIME system upon its validation)

[

]^{a,c}

b. "how this may impact the spacer grid structural behavior"

The grid impact occurs on the grid side surfaces. For example, a grid is impacted in the X direction as shown in Figure 4. The impact force is shared by the whole column A (from cell A1 to cell A17) and is transferred to the whole grid through all columns (from Column A to Column Q). Therefore, the cell gap size differences in a grid would have a small impact on the overall structural behavior of a spacer grid.

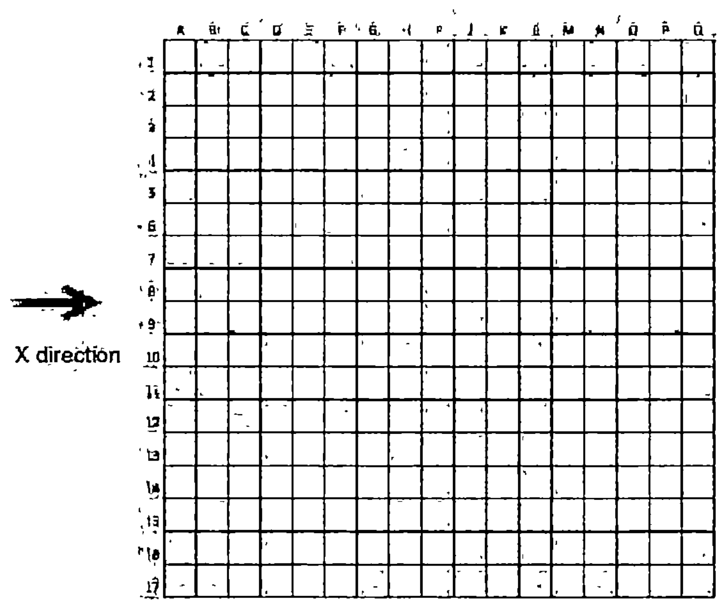

Figure 4. Example of Grid Impact

The [ ]^{a,c} cell gaps for simulating the EOL grids are also varied across grid locations. The pre-sized cells realistically represent the measured cell gap characteristics from the PIE data, such as random distribution, gap size range, etc. [ ]^{a,c}

[

.]a, c

a, c

Figure 5. Measured [ ]a, c

[

.]a, c

Overall, the grid impact between two fuel assemblies and a fuel assembly to a baffle plate occur on the side surface of the grid. The impact surface transfers the impact force through all the grid straps (which are parallel and perpendicular to the impact direction) The average gap size in each column and each row for a simulated EOL grid is similar. Therefore, [

.]$^{a, c}$

RAI 3

Section 2.1 of the LTR presents PIE data from selected fuel assemblies and an analysis approach that can be used to determine a target average cell size for the simulated EOL spacer grids. This approach is intended for use with any fuel assembly design, but no specific guidance is provided on how the PIE data set should be characterized for a given fuel assembly design. Please provide guidance on the expectations for what would constitute an acceptably robust set of PIE data for the purpose of establishing a bounding target average cell size for all fuel assemblies of the specified design type.

Response to RAI 3

The grid target cell size is determined based on the PIE data using a statistical method. For example, the grid target cell size for [ ]$^{a, c}$ fuel assembly is determined by the following steps:

1 [

.]$^{a, c}$

2 Calculate the upper 95% confidence limit for the true mean in order to account for the scatter in the database. The upper 95% confidence limit is calculated based on the statistical formula given below

$$MidGrid_{upper95} = MidGrid_{avg} + \frac{STD \times T_{n-1}}{\sqrt{N}}$$

Where:

MidGrid$_{upper95}$ —— Upper 95 confidence mean of the grid cell gap size from the PIE data

MidGrid$_{avg}$ —— Average grid cell gap size from the PIE data

STD —— Standard deviation of the grid cell gap size from the PIE data

T —— Student T value determined by the sample size

*** This record was final approved on 5/14/2018 10 48 53 AM (This statement was added by the PRIME system upon its validation)

N —— Sample size

n-1 —— Sample size minus one

[

]a, c

3. [

.]a, c

[

]a, c

Figure 6. [

]a, c

To ensure that the simulated EOL grids meet the target cell gap value, the average cell gaps should be higher than the target cell gap value. For an additional conservatism, the lower 95% confidence limit on the true mean of tested grid cell gaps was confirmed to be higher than the target value. The simulated average and lower 95% cell gaps for a [

.]^{a,c}

In general, the effect of the sample size is incorporated into the statistical method through the "Student T" value in the formula above. Smaller sample sizes will have a larger Student T value. For example, [

.]^{a,c}

Table 2. Example of the Gap Size Target Value Utilizing

[ ]^{a,c}

a, c

[ ]a, c

Figure 7. Sample [ ]a, c

Based on the discussions above, [ ]a, c is sufficient.

References:

1. [.]a, c

PWROG
PWR Owners Group

Program Management Office
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

PWROG-16043-P, Revision 2
Project Number 99902037

March 27, 2018

OG-18-62

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: PWR Owners Group
Transmittal of the Response to Request for Additional Information, RAIs 4 and 5 Associated with PWROG-16043, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs", PA-ASC-1169

References:

1. Letter OG-17-12, Submittal of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," PA-ASC-1169R2, dated February 1, 2017

2. NRC Letter of Acceptance for Review of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," dated June 20, 2017

3. Email from the NRC (Benney) to the PWROG (Holderbaum), Request for Additional Information, RAIs 1-6, RE: PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," dated January 31, 2018

On February 1, 2017, in accordance with the Nuclear Regulatory Commission (NRC) Topical Report (TR) program for review and acceptance, the Pressurized Water Reactor Owners Group (PWROG) requested formal NRC review and approval of PWROG-16043-P, Revision 2 for referencing in regulatory actions (Reference 1). The NRC Staff has determined that additional information is needed to complete the review per letter dated January 31, 2018 (Reference 3).

U.S. Nuclear Regulatory Commission March 27, 2018
OG-18-62 Page 2 of 3

Enclosure 1 to this letter provides a response to NRC RAIs 4 and 5 (Reference 3) associated with PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs,"

Also enclosed are the Westinghouse Application for Withholding Proprietary Information from Public Disclosure, CAW-18-4722, 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 item listed above or the supporting Westinghouse Affidavit should reference CAW-18-4708 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.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Executive Director
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, PA 16066

If you have any questions, please do not hesitate to contact me at (805) 545-4328 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855.

Sincerely yours,

Ken J Schrader

Ken Schrader, COO & Chairman
PWR Owners Group

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U.S. Nuclear Regulatory Commission March 27, 2018
OG-18-62 Page 3 of 3

JKS:am

cc: PWROG Analysis Committee (Participants of PA-ASC-1169)
PWROG PMO
PWROG Steering and Management Committee
J Andrachek, Westinghouse
K. Lasswell, Westinghouse
J. Sinegar, Westinghouse
B. Benney, US NRC

Enclosure 1 PE-18-24-P/NP, Attachment 1, "RAIs 4 and 5 Responses for PWROG-16043 Revision 2" (PA-ASC-1169)

Enclosure 2. Affidavit for Withholding, CAW-184722 (Non-Proprietary) with accompanying Affidavit, Proprietary Information Notice and Copyright Notice

Westinghouse Non-Proprietary Class 3

Westinghouse

From: Roger Yong Lu
Phone: (803) 647-3426
e-mail: lur@westinghouse.com

Our Ref: PE-18-24-NP
Date: March 15, 2018

Subject: **Response to PWROG Topical Report PWROG-16043-P RAIs 4 and 5**

To: James P. Molkenthin Jill G. Sinegar James D. Andrachek
cc: PWROG

Attached are the responses to RAIs 4 and 5 related to the PWROG Topical Report PWROG-16043-P. Proprietary information is shown in brackets. Questions or comments should be directed to the undersigned.

Author: Roger Y. Lu*
PWR Fuel Technology

Verifier: Jane X. Jiang*†
Thermal-Hydraulic and Seismic Engineering

Approver: Kevin T. Lasswell*, Manager
Thermal-Hydraulic and Seismic Engineering

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† Three Pass Verification (3PV) was used to verify this document, as demonstrated in the Design Document Verification Checklist which is attached to this document in EDMS.

PE-18-24-NP,
Attachment 1
March 15, 2018
Page 1 of 1

RAI 4

The LTR states that beyond the use of simulated EOL grids, no modification was made to the NRC-approved testing and analysis methodologies documented in WCAP-9401-P-A and CENPD-178OP, Rev. 1-P. Therefore, NRC approval is not being sought for anything beyond the proposed use of simulated EOL grids to determine the allowable grid impact strength. In order to verify that NRC review and approval beyond the limited scope described in the LTR is not necessary, the NRC staff requests the following clarification:

4. In Section 2 2 of the LTR, the allowable grid impact strength for CE and Westinghouse fuel are discussed as [

]$^{a, c}$

Response to RAI 4

[

]$^{a, c}$

RAI 5

A new methodology is being proposed for Westinghouse and CE fuel to credit flowing water damping in mitigation of the degradation in fuel mechanic behavior due to EOL effects on the spacer grids. This methodology is proposed as an option for use in lieu of the still water damping credited in the previously approved methodologies. In order to fully understand how the proposed methodology is intended to conservatively capture the impact of flowing water on fuel assembly vibrations, the NRC staff requests the following information:

5. Section 4 of the LTR discusses application of flowing water damping for EOL conditions. Please clarify whether the EOL conditions, with flowing water damping, will be considered to bound BOL conditions, or if BOL conditions will continue to be analyzed separately with the existing still water damping methodology. If the EOL condition analysis is intended to bound BOL conditions, please provide information justifying this conclusion.

Response to RAI 5

The EOL conditions that considered flowing water damping do not bound BOL conditions. The BOL conditions will continue to be analyzed separately with the existing still water damping methodology.

*** This record was final approved on 3/25/2018 11 17:51 AM (This statement was added by the PRIME system upon its validation)

June 20, 2017

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive, Suite 380
Cranberry Township, PA 16066

SUBJECT: ACCEPTANCE FOR REVIEW OF THE PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-16043, "PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09: IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS" (CAC NO. MF9280)

Dear Mr. Nowinowski:

By letter dated February 1, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17039B050), the Pressurized Water Reactor Owners Group (PWROG) submitted Topical Report (TR) PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

The NRC staff has found that the material presented is sufficient to begin our review. The NRC staff expects to issue its request for additional information by March 30, 2018, and issue its draft safety evaluation (SE) by September 3, 2018. This schedule information takes in consideration the NRC's current review priorities and available technical resources and may be subject to change. If modifications to these dates are deemed necessary, we will provide appropriate updates to this information. The review schedule milestones were discussed and agreed upon in a telephone conference between PWROG Project Manager, Chad Holderbaum, and the NRC staff on June 14, 2017.

Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review.

Section 1.4 of PWROG-16043-P specifies the limited scope review being requested by PWROG. This section clearly states that this topical does not "revise and or modify the current NRC-approved grid and fuel assembly test methods, or the fuel assembly seismic and loss-of-coolant accident analysis methodologies, processes and codes." Section 1.3 of PWROG-16043-P goes on to state that this topical report "does not supercede the NRC-approved TRs WCAP-9401-P-A (Reference 1-3) and CENPD-178-P, Rev. 1-P (Reference 1.4)." The NRC staff understands the limited scope review being requested and does not intend to expand its review into the underlying, legacy seismic methods within WCAP-9401-P-A or CENPD-178-P, Rev. 1-P.
However, issues identified within these legacy methods during recent and ongoing new reactor reviews (i.e., AP1000 and APR1400) may need to be addressed prior to use of the revised end of life fuel characteristics and damping coefficients in PWROG-16043-P.

W. A. Nowinowski - 2 -

As a result, the staff's safety evaluation may include a limitation and condition defining issues with the legacy methods, which need to be resolved prior to use of PWROG-16043-P.

As with all TRs, the SE will be reviewed by the NRC's Office of the General Counsel (OGC) to determine whether it falls within the scope of the Congressional Review Act (CRA) During the course of this review, OGC considers whether any endorsement or acceptance of a TR by the NRC amounts to a rule as defined in the CRA. If this initial review concludes that the SE, with its accompanying TR, may be a rule, the NRC will forward the package to the Office of Management and Budget (OMB) for further review and consideration. Any review by OMB would impact the schedule for the issuance of the final SE. If you have questions regarding this matter, please contact Brian Benney at (301) 415-2767.

Sincerely,

/RA/

Dennis C Morey, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

W. A. Nowinowski -3-

SUBJECT: ACCEPTANCE FOR REVIEW OF THE PRESSURIZED WATER REACTOR OWNERS GROUP TOPICAL REPORT PWROG-16043, "PWROG PROGRAM TO ADDRESS NRC INFORMATION NOTICE 2012-09: IRRADIATION EFFECTS ON FUEL ASSEMBLY SPACER GRID CRUSH STRENGTH FOR WESTINGHOUSE AND CE PWR FUEL DESIGNS" (CAC NO. MF9280) DATED: JUNE 20, 2017

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ADAMS Accession No.: ML17123A125; *concurred via e-mail NRR-106

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NAME	PClifford	DMorey	
DATE	6/1/17	6/20/17	

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PWROG
PWR Owners Group

Program Management Office
20 International Drive
Windsor, Connecticut 06095

February 1, 2017

OG-17-12

Document Control Desk
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: PWR Owners Group
Submittal of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," PA-ASC-1169R2

Reference: **NRC Information Notice 2012-09, "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength," dated June 28, 2012.**

The purpose of this letter is to submit Pressurized Water Reactor Owners Group (PWROG) Topical Report (TR), PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs," in accordance with the Nuclear Regulatory Commission (NRC) TR program for review and acceptance for referencing in regulatory actions. PWROG-16043-P, Revision 2 is provided in Enclosure 1

PWROG-16043-P, Revision 2 addresses the issue identified in NRC Information Notice 2012-09 by applying the approach that was used to address the End of Life (EOL) effects for the AP1000®[1] Core Reference Report APP-GW-GLR-153, Rev. 1, "AP1000 Core Reference Report." PWROG-16043-P, Revision 2 discusses the applicability for determining fuel assembly characteristics and damping coefficients at EOL conditions and the aspects for which NRC approval is requested. PWROG-16043-P, Revision 2 does not revise and/or modify the current grid and fuel assembly test methods, or the fuel assembly seismic and LOCA analysis methodologies, processes and codes that were previously approved by NRC.

Enclosure 3, contains Westinghouse authorization letter CAW-17-4530, the accompanying affidavit, Proprietary Information Notice, and Copyright Notice.

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[1] **AP1000 and CE16NGF** are 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.

PWROG-16043-P, Revision 2, contains information proprietary to Westinghouse Electric Company LLC, therefore 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.

Accordingly, it is respectfully requested that this 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 information identified above or the supporting Westinghouse affidavit should reference CAW-17-4530, and should be addressed to Mr. J. A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company LLC, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania, 16066.

TR Classification: As discussed above, this TR addresses the issue associated with the irradiation effects on fuel assembly spacer grid strength identified in NRC Information Notice 2012-09, via a generic licensing action, that will be used for evaluating the structural integrity of fuel assemblies under faulted condition loads (seismic and LOCA) for Westinghouse and CE fuel designs at EOL conditions, on a plant-specific basis.

Specialized Resource Availability: This TR is being submitted to the NRC for review and approval so that the NRC approved version can be utilized for performing plant-specific evaluations of the structural integrity of fuel assemblies under faulted condition loads (seismic and LOCA) for Westinghouse and CE fuel designs at EOL conditions NRC approval of the generic TR will provide a common approach that will be utilized to address the EOL effects on fuel assembly space grid strength in fuel assembly structural integrity evaluations

This letter transmits four copies of PWROG-16043-P, Revision 2 (Enclosure 1) and one copy of PWROG-16043-NP, Revision 2 (Enclosure 2)

Applicability: This TR is applicable to the Westinghouse and CE Nuclear Steam Supply System (NSSS) plants that are participating in the PWROG program, PA-ASC-1169R2, that developed this TR

Request for Review Fee Waiver

The PWROG will be requesting that a fee waiver be considered for the NRC review of PWROG-16043-P, Revision 2 pursuant to the provisions of 10 CFR 170.11(a)(1)(ii)(A). PWROG-16043-P provides a common approach that will be utilized to address the EOL effects on fuel assembly space grid strength in fuel assembly structural integrity evaluations. NRC approval of the TR will ensure that the EOL effects on fuel assembly space grid strength in fuel assembly structural integrity evaluations are considered in these evaluations Therefore the review of this TR will support ongoing

U.S. Nuclear Regulatory Commission February 1, 2017
OG-17-12 Page 3 of 4

NRC generic regulatory improvements/efforts associated with the issue of EOL effects on fuel assembly space grid strength in fuel assembly structural integrity evaluations.

During the fee waiver decision period, the PWROG respectfully requests the NRC Staff to perform its acceptance review of PWROG-16043-P, Revision 2 The PWROG will assume the responsibility of payment of the NRC review fees accrued both during the acceptance review, and during the review, if the fee waiver is not approved.

NRC Review Schedule

The PWROG requests that the NRC complete their review of the TR by August 2018.

Correspondence related to this transmittal should be addressed to:

Mr. W. Anthony Nowinowski, Program Manager
PWR Owners Group, Program Management Office
Westinghouse Electric Company
1000 Westinghouse Drive
Suite 380
Cranberry Township, Pennsylvania, 16066

If you have any questions, please do not hesitate to contact me at (205) 992-7037 or Mr. W. Anthony Nowinowski, Program Manager of the PWR Owners Group, Program Management Office at (412) 374-6855

Sincerely yours,

Norman J. Stringfellow

Jack Stringfellow, Chief Operating Officer and Chairman
PWR Owners Group

NJS.WAN

Enclosures 1 and 2. Four copies of PWROG-16043-P, Revision 2, "PWROG Program to Address NRC Information Notice 2012-09: "Irradiation Effects on Fuel Assembly Spacer Grid Crush Strength" for Westinghouse and CE PWR Fuel Designs" (Proprietary) and one copy of PWROG-16043-NP, Revision 2

Enclosure 2: One copy of the Application for Withholding, CAW-17-4530 (Non-proprietary) with the accompanying affidavit, Proprietary Information Notice and Copyright Notice

U.S. Nuclear Regulatory Commission
OG-17-12

February 1, 2017
Page 4 of 4

cc: PWROG Management Committee
PWROG Analysis Committee
PWROG Steering Committee
PWROG Licensing Committee
PWROG PMO
J. Gresham, Westinghouse
J. Andrachek, Westinghouse
J. Moorehead, Westinghouse
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