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THE U.S. NUCLEAR REGULATORY COMMISSION STAFF FEEDBACK REGARDING ARC CLEAN TECHNOLOGY, INC. "WHITE PAPER ON FUEL QUALIFICATION," REVISION 0.0 (EPID NO. L-2023-LRO-0071)

SPONSOR INFORMATION

Sponsor:	ARC Clean Technology (ARC)
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Purpose of the White Paper: By letter dated October 12, 2023, ARC Clean Technology, Inc. (ARC) submitted for the U.S. Nuclear Regulatory Commission (NRC) staff's review the white paper (WP) entitled "White Paper on Fuel Qualification," Revision 0.0. The WP summarizes ARC's plan to qualify fuel to support operation of the ARC-100 reactor.

ARC indicated that the purpose of the WP is to elicit the NRC's views on whether ARC's fuel qualification plans are sufficient to support an eventual license application.

Action Request: ARC requested that the NRC staff review the WP and provide feedback regarding topics for which additional discussion or consideration may be beneficial. ARC also provided specific questions that it would like the NRC staff to address in its feedback (ML24213A207), as listed in Section 1 of this document, "ARC Specified Feedback."

Guidance

Guidance on fuel qualification for non-light-water reactors is provided in NUREG-2246, "Fuel Qualification for Advanced Reactors" (ML22063A131). This NUREG builds on guidance for light-water reactors contained in NUREG-0800, Section 4.2, "Fuel System Design," Revision 3 (ML070740002). Additional relevant technical guidance is available in NUREG/CR-7305, "Metal Fuel Qualification: Fuel Assessment Using NRC NUREG-2246, 'Fuel Qualification for Advanced Reactors'" (ML23214A065) which is a generic assessment of uranium-zirconium alloy metallic fuel following the NUREG-2246 process. While NUREG/CR-7305 assumes certain characteristics of the fuel for the purposes of its assessment, most of the discussion in the report is relevant to the ARC-100 fuel.

Enclosure

FEEDBACK AND OBSERVATIONS

The feedback and observations on this WP are preliminary and subject to change. The feedback and observations are not regulatory findings on any specific licensing matter and are not official agency positions. The feedback and observations on this WP are also not intended to be comprehensive; a lack of feedback or observations should not be interpreted as NRC staff agreement with ARC's position.

ARC indicated that some of the work described in the WP is still in progress; the NRC staff's feedback is based only on the information provided in the WP.

The NRC staff feedback is organized into three sections as follows:

1. ARC-100 Specified Feedback: Response of the NRC staff to address specific questions posed by ARC.

2. Specific Observations: Specific technical observations relevant to information found in the WP and the performance of the fuel system.

3. General Observations: Observations regarding the overall structure and content.

SECTION 1

ARC-100 SPECIFIED FEEDBACK

1. ***The completeness of the elements of the proposed program, i.e., are there additional steps that ARC should undertake?***

   The NRC staff identified that additional verification and validation of fuel performance would be needed to support a fuel qualification licensing submittal. Several ARC-100 fuel design parameters (e.g., width, length, in-reactor time, fluence, and peak cladding exposure) substantially deviate from those employed in Experimental Breeder Reactor-II (EBR-II) and Fast Flux Test Facility (FFTF) as shown in table E-1, "Enveloping Design Parameters for Metal Fuel from Existing Database," of the WP. The WP acknowledges that design or operating parameter deviations could affect the overall performance of the fuel system. However, simulations used to demonstrate that deviation effects are minimal rely on EBR-II or FFTF material properties and behavioral models resulting in increased uncertainty that simulation results accurately reflect ARC-100 fuel performance.

   A draft report from a previous study by Brookhaven National Laboratory (BNL), "Assessment of the Adequacy of Metallic Fuel Qualification to Support Licensing of Small Modular Sodium-Cooled Fast Spectrum Reactors (4S, ARC100, PRISM)" (reference 88 in the WP), also emphasizes this fact, stating:

   > *… it is recommended that unless the proposed reactor has the same dimensions and operating parameters as the EBR-II, additional work will be*
   >
   > *required to make a convincing safety case. The amount of additional work required will depend on the magnitude of the deviation from the EBR-II configuration. If the proposed [sodium fast reactor (SFR)] had a configuration*

> *covered by the design parameters shown in Table 6.2 ["Enveloping Design Parameters for Metal Fuel from the Existing Database"], a "modest" amount of work would be required. This is because the fuel composition is the same as that used in the EBR-II, and the primary differences are modest dimensional changes. However, deviations beyond those shown in Table 6.2 that would not allow extrapolation with the requisite confidence would imply a "significant" amount of work required to license the fuel …*
>
> *…The experimental data available from EBR-II and FFTF provide relevant data for qualifying metallic fuel for the 4S, PRISM, and ARC-100 SMRs and supporting the licensing process. However, the differences noted above in Table 6.1 ["Comparison of Characteristics of Metallic Fuels"] make it difficult to argue that the EBR-II and FFTF data address the "prototypic" objective. This is especially true for the 4S and ARC-100 reactors…*

The NRC staff agrees with these assessments. Correlations, material properties, and behavioral models that were developed from the operational experience of EBR-II and FFTF may not be sufficient or applicable when the parameters, either operational or geometric, change considerably. If ARC intends to extensively leverage modeling and simulation tools for the quantification of key performance parameters such as temperature, stress, and strain, it may necessitate a more comprehensive collection of material properties and behavioral models. This should be supplemented by experimental verifications to ensure accuracy and reliability.

2. ***Have we provided a reasonably complete list of all the experimental and operational data that is or can be used to justify the proposed approach?***

The WP provides a reasonably complete list of relevant experimental and operational data. The NRC staff understands that ARC intends to rely on the operational experience from EBR-II and FFTF and supplement the data with experiments, if possible, while in operation. Additional pre-application engagement is suggested if ARC identifies that data during operation is needed to support fuel qualification. While the NRC staff agrees that EBR-II and FFTF operational experience is relevant to the qualification of ARC-100 fuel; additional verification, validation and justification would be needed in a licensing submittal to reduce uncertainty in the impacts of fuel design parameter deviations.

For instance, Section 5.4, "Design Gaps," of the WP acknowledges that there is a gap in the data for HT-9 swelling **[[**
]]. However, Section 5.4 does not discuss that the residence time at elevated temperatures **[[**
]]. Even if the cladding experienced the same fluence as EBR-II and FFTF, the longer time under elevated temperatures may lead to a different creep response.

3. ***Do the two conformance matrices (to NUREG-2246 and NUREG/CR-7305) clearly identify where the ARC 100 fuel planned operational performance falls outside the range of existing operational and experimental data?***

The NRC staff considers NUREG-2246 and NUREG/CR-7305 to provide sufficient information to identify where the proposed operational parameters of ARC-100 fuel falls outside the range of existing operational and experimental data. The NRC staff note that the

BNL report, cited in item 1 above, also provides information germane to the identification of fuel design and operational parameters that are enveloped by existing operational and experimental data.

4. ***Have we properly recognized the limitation of that data (i.e., gaps) that underpins the computer programs (BISON and MFUEL module of SAS4A/SASSYS-1) used to determine the fuel performance over time, and described the step we propose taking to compensate for those uncertainties (i.e., fill the gaps)?***

   The WP adequately highlights limitations in the data that underpin the computer programs used to determine fuel performance over time. The results provided by BISON are highly dependent on the material properties and behavioral models used in simulations.

   The NRC staff recognizes that BISON can provide accurate results if appropriate inputs are used. However, use of material properties or behavioral models that do not adequately reflect fuel design parameters or operating regimes can mask errors in simulation results. Additional verification and validation would be needed in a licensing submittal to show that the behavioral models and material properties used as inputs to BISON are adequate given the ARC-100 novel fuel design and operating regimes.

   For example, consider the modulus or irradiation creep of irradiated fuel. If the correlation is from the *Metallic Fuels Handbook* (reference 21 in the WP), the handbook indicates that creep data are scarce. The proposed correlation is therefore a best-estimate correlation based on assumptions and other fuel types such as UO_2 without characterization of associated uncertainties. Similar issues arise with the cladding properties, such as irradiation-enhanced creep of the cladding. If the correlation is based on data from EBR-II and FFTF, which had significantly shorter in-reactor time, the correlation may not be able to capture the creep response of the cladding for the proposed ARC fuel.

5. ***Have we sufficiently justified that the acceptance criteria proposed for the design of the fuel are appropriately conservative and, with justification, could be exceeded? For example, are peak strain limits for thermal and irradiation creep, which are not directly measurable during operation, acceptable limits?***

   The acceptance criteria proposed are generally consistent with the acceptance criteria specified for EBR-II Mark-V fuel in ANL-NSE-1, "Safety Analysis and Technical Basis for Establishing an Interim Burnup Limit for Mark-V and Mark-VA Fueled Subassemblies in EBR-II." Because the Mark-V fuel acceptance criteria were demonstrated to be conservative with respect to predicting fuel failure, it seems reasonable to use similar criteria for ARC-100 fuel. However, given the feedback provided in items 1-4 above, the NRC staff cautions that the numerical values for the acceptance criteria may need further justification due to potential non-applicability of the data that underlie the correlations or calculations performed for the various acceptance criteria.

   In addition, there was limited information on how differences between ARC-100 fuel and historical metallic fuel may affect the identification and assessment of failure modes. Specifically, failure is generally not solely defined based on peak strain. While strain provides some information, it may not be sufficient alone, especially for a material that loses its ductility. In such cases, peak stresses and other parameters such as fracture toughness should also be considered.

Furthermore, strain should not be limited to thermal and irradiation creep (i.e., time-dependent plastic strains); combined deformations, including strains from fabrication (i.e., residual strains), should also be considered. If a strain-based failure mode analysis is to be conducted, then it is difficult to justify that setting limits solely on thermal creep or irradiation creep alone would be sufficient in determining structural integrity. Total strain should be defined as the sum of initial, elastic, and inelastic strains. Elastic strain is the sum of elastic strain and thermal expansion strain. Inelastic strain is the sum of time-independent and time-dependent strain, including plastic strain (i.e., J2 plasticity), irradiation growth, and irradiation-enhanced creep. Each of these could be less than the fracture strength of the cladding at a certain dose and temperature, but any combination could challenge the integrity of the fuel elements.

Dynamic effects are another consideration. The strength of the material could be highly dependent on the strain rate. For example, a 2% peak strain could be adequate if the material experiences straining over the long term. However, the material may be highly susceptible to rupture if the straining occurs in a short period of time. The state-of-practice approach for assessing failure modes is to identify possible failure modes first to determine the nature of the loading. Stresses, strains, and toughness (e.g. fracture toughness) should be considered together to make an assessment with reasonable accuracy.

The importance of considering combined strains is highlighted in the design rules for metallic components in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, Division (Div) 5, "High Temperature Reactors." The code states that maximum accumulated inelastic strain over any cross-section of the wall thickness of the structure shall not exceed: 1% strain averaged through the thickness, 2% strain at the surface, and 5% localized strains at any point, while those limits are reduced by factor of two (x2) for the weldments.

Therefore, while it is reasonable to define limits for irradiation-enhanced creep and thermal creep, meeting these alone may not be sufficient to ensure the integrity of the cladding. Complete loading combinations should also be considered. For example, a fuel assembly may experience bowing, which could induce plastic strain. There may be plastic strains resulting from fabrication. Therefore, the **[[**

]].

Regardless of the failure criteria used, the NRC staff will review the details of such evaluations and how those evaluations will be made to assess their accuracies.

6. ***Have we identified both sufficient analytical and testing steps to compensate for those limitations, so there can be good confidence the fuel will behave as predicted?***

   As described above, further verification, validation and assurance would be needed in a licensing submittal to ensure that the correlations, material properties, and behavioral models derived from EBR-II and FFTF are applicable to the design of the ARC-100.

   a. ***More specifically, have we outlined a sufficiently exhaustive plan of analytical investigation of the effects of uncertainties, e.g., dimensions, material properties, temperature prediction, adequacy of correlation employed in the codes, on the fuel behavior prediction?***

The NRC staff would need additional information on material properties to complete an adequate performance assessment; specifically, additional information on irradiated material properties of the fuel system is needed. Items for further evaluation include which properties are available, their accuracies, and how they are applicable to the ARC-100 fuel design. Based on the result of a material property assessment study, parametric sensitivity studies should be conducted to observe the effects of a single material property-related parameter on the performance prediction.

b. ***Have we adequately identified the time and conditions for which the behavior of the fuel might start to deviate from the prediction?***

Based solely on the material properties and behavioral models used in simulations, yes. However, as described above, additional justification would be needed to demonstrate that the behavioral models from previous operational experience are applicable to the ARC-100 fuel geometry. If the operational and geometrical parameters were the same or closely comparable, the proposed approach may adequately assess when the behavior of the fuel deviates from the prediction.

c. ***Is the proposed online operational inspection and testing program sufficiently explained, so there can be confidence it will provide timely information on the fuel behavior?***

The general description of the inspection and testing program is reasonable. However, more frequent extraction and evaluation may be warranted due differences in ARC-100 fuel design parameters compared with historical precedence. Additionally, more details on the testing program are needed regarding what properties or measurements would be assessed.

d. ***Is the proposed periodic extraction of fuel assemblies and pins for confirmation of the fuel behavior a sufficient complement to the analytical and online testing program?***

Proposed extractions to confirm the predictions are necessary, but additional information is needed to assess sufficiency. More frequent extractions may be needed to verify that the fuel behavior is not deviating from the predicted behavior.

7. ***Is the proposed fabrication program adequate to ensure the as fabricated fuel meets all the design specifications?***

The NRC staff recognizes that ARC has considered many aspects of fuel fabrication. As summarized in the BNL study (reference 88 in the WP), the fuel fabrication process should be "equivalent" to that of fuel irradiated in EBR-II/FFTF to support the argument that those irradiations are prototypic of ARC-100 fuel.

A key aspect of a fabrication program is ensuring the fabrication method produces a product consistently within specifications, which can be achieved through a rigorous quality control process. This is crucial since some parameters could have impacts on the overall performance of the fuel system. Any deviations from those parameters may lead to different fuel behavior. For instance, pre-irradiation grain size and texturing in the fuel could affect the swelling and creep of the fuel zone, which may have implications for the forces exerted on the cladding walls. Similarly, different post-fabrication residual stresses in the cladding could influence its strength. Another example is the welding procedure and welding parameters, which become especially important due to historical failures occurring at or near welds.

Fabrication parameters that could affect the in-reactor performance of the fuel system should be identified. These influential parameters should be monitored and controlled to ensure post-fabrication consistency. This is necessary to ensure a "predictable and stable behavior" of the fuel system for comparable irradiation settings.

SECTION 2

SPECIFIC OBSERVATIONS

1. The NRC staff notes that including the enrichment level (i.e., EBR-II >60%, ARC-100 <20%), in table E-1 of the WP would make the table more complete, as enrichment is another parameter to consider. It is not clear to the NRC staff based on information provided in the WP how the lower enrichment proposed for ARC-100 fuel impacts overall fuel behavior. For example, if the fission rate is different, it may impact composition, zirconium migration, fuel restructuring, porosity characteristics, fuel swelling, irradiation-induced creep, etc. Considering that EBR-II used 60-70 percent enriched fuel, additional information would be needed in a licensing submittal for the NRC staff to adequately evaluate if the behavioral models of the fuel from EBR-II are still applicable.

2. Page 36 of the WP, item 2.3, “Fuel Performance Envelope – Radial Strain and Deformation” emphasizes fuel thermal creep. ARC indicated that the models were developed by using a limited data set. The NRC staff will need more information on how the irradiation-induced creep correlation was developed. This is an important parameter, along with the fuel swelling and pressure; the irradiation creep of the fuel zone will have implications on the stresses and strains exerted on the cladding.

3. On Page 41 of the WP, item 4.2, “Fuel Performance Modeling,” ARC indicated that MFUEL was used to confirm the results from BISON. The NRC staff identified similar statements in various sections of this WP. Although code-to-code comparisons can be informative, the context in which they are used is important. Comparing the results from one software package with the results from another is not a reliable validation method if both software packages are using the same correlation; matching the results only implies that both packages are using that correlation correctly. It does not confirm the accuracy of the correlations that those software packages are utilizing. The correlations used to obtain simulation results must be validated by an alternative means, such as comparison against independent data (i.e., data that was not used to develop the correlation).

4. From Figure 3.1, “Fuel Qualification Program Elements” of the WP, it is not clear to the NRC staff what steps are taken if there is an identified data gap and the sensitivity studies show that there is a significant impact on performance predictions. Additional clarification is needed regarding whether actual testing and property measurements will be included in a licensing submission.

   It appears to the NRC staff that the “Testing” component of the flowchart are standalone, implying that “irradiated test” program does not take any inputs from any other activities. While Box 14, “Design Out-of-Pile Tests” receives some input from Box 8, “Are there gaps in the data? Does data require extrapolation for use in ARC-100", it seems "Box 16, Perform In-Pile Test" is standalone is not receiving any input from other activities. If the NRC staff understood the flowchart correctly, starting from Box 8, if data gaps in irradiated properties are identified, it is difficult to conclude that increasing the safety factor in calculations alone would ensure safety margins are met, especially when many mechanical properties for the irradiated fuel and cladding are either not known or have large uncertainties.

5. On Page 46 of the WP, it is not clear to the NRC staff if the following statement means the peak fuel temperature is yet to be established: "… limits in steady state peak fuel and cladding 2σ temperatures, have been established to be **[[ ]]**…"

   In addition, the note on table 3-1, "Design Criteria for ARC100 Fuel," of the WP and the discussion of Criterion 3.1.5 on Page 51 of the WP indicate that **[[ ]]** was used as the maximum cladding stress in past qualification, but one-half of the yield strength at temperature would be a more appropriate limit. The justification and applicability of the proposed limit are not clear to the NRC staff.

   Table 3-1 and the discussion of Criterion 3.1.5 also indicate that the peak stresses will be checked against the yield strength at a temperature. The discussion of Criterion 3.1.5 also mentions that the HT9 yield strength at high temperature is a function of strain rate and irradiation history. Therefore, the comparison should also consider the stress-strain for different irradiation times; the WP does not appear to be consistent about whether these parameters will be part of the consideration.

   [[ ]].

6. Page 48 of the WP states, "Permanent strain consists of a volumetric swelling strain and in-reactor thermal creep and irradiation creep strain."

   If HT9 is being referred to, it is not clear to the NRC staff why ARC does not consider plastic strains (J2 plasticity) as a permanent strain. For instance, if the fuel deforms the cladding in the reactor at an elevated temperature, the total strain in the cladding should be the sum of elastic and inelastic strains, where the inelastic strains include plastic, swelling, creep and initial strains.

7. The discussion of Criterion 3.1.7 on page 52 of the WP indicates that ARC intends to perform CDF (Cumulative Damage Function) evaluations based on the previously performed burst tests. The approach seems reasonable to the NRC staff, assuming that the irradiated pins are comparable with the dpa and fluence targeted by ARC. Irradiated cladding from EBR-II may not be representative as those experiments had lower dpa, fluence and shorter high temperature exposure.

8. Page 54 of the WP states, "Structural requirements can be defined once the service classification of the assembly is determined. Following the guidance of the ASME Code Section III, subsection NB, the assembly will be subjected to four service levels, A, B, C, and D."

Careful assessment is necessary to ensure that the appropriate ASME code case is applied correctly. The NRC staff notes that Section III, Div 1 has a temperature limit for applicability. ASME indicates that Div 1 is applicable up to 371 °C (700 °F) for ferritic steels and 427 °C (800 °F) for austenitic steels. ARC should ensure that the temperature of the fuel assembly would fall into the temperature range covered by Div 1. For instance, **[[ ]]**. Similar verification should be done for other parts such as ductwork, fittings, etc.

9. The NRC staff notes that additional information would be expected in a future licensing submittal on welding parameters described in Section 3.2.4, "Special Requirements for Weldments," of the WP (e.g., such as pressure, current, hold time, and cooling rate). The NRC staff would also need to review detailed evaluations, not only for the weldment seams but also the heat-affected zone. These detailed evaluations are particularly important since 16 of the total 24 failures that were observed in Mark III/IV fuel at EBR-II occurred at weldments or at the heat-affected zone due to fabrication defects.

10. Page 57 of the WP indicates that modeling and simulation activities were, or will be, performed using axisymmetric models. The NRC staff notes that an axisymmetric model may not be sufficient to estimate the total strains of the cladding. For instance, if the fuel assembly is bowed (i.e., deviated from its neutral axis), an axisymmetric model may not accurately account for the plastic strains in the cladding. The NRC staff will need more information about ARC's modeling methodology as part of a future submittal to accurately evaluate modeling and simulation results.

11. Section 5.3.1, "BISON Predictions," of the WP provides BISON results without sufficient assessment of their accuracies. As indicated under item 3 of this feedback section, running the simulations using another software and obtaining the same results is not necessarily an appropriate method for validation. Ideally, all the correlations and material properties that were used (in BISON, for this instance) and their valid ranges should be provided.

    Acknowledging the fact that ARC's geometry and operational parameters are different from those in EBR-II and FFTF, the applicability of the correlations and material properties used in simulations should be examined carefully. Using a correlation or a material property that is not representative of the ARC design could mask large errors.

    For instance, table 5-6, "End-of-life design criteria characteristics for ARC-100 fuel pin," of the WP indicates that circumferential stress was estimated as **[[ ]].** Based on the results, ARC predicts zero failures. However, when irradiated properties for fuel and cladding materials are absent or contain large uncertainties, the accuracy of results coming from the software is potentially unreliable.

12. In Figure 5.8, "**[[ ]]**" of the WP, it is not clear to the NRC staff why the predicted temperature profile oscillates for the elevations between **[[ ]]**. A similar response is seen in Figure 5.13, "**[[ ]]**" Figure 5.16, "**[[ ]]**," Figure 5.17, "**[[**

]]" and Figure 5.20, "**[[**
]]".

13. Related to Section 5.4, "Design Gaps," of the WP, the NRC staff notes that a comprehensive study on irradiated material properties of the fuel system is needed. Such a study may include a list of material properties; behavioral models; and currently available property measurements, their respective uncertainties, and an assessment of their applicability to the ARC design.

    Section 5.4.1, "Swelling data for HT9," describes the use of irradiation with heavy ions to mimic the swelling of fuel and cladding due to fast neutron irradiation over a shorter length of time. The NRC staff has not reviewed the viability of these studies and notes that the technique should be approached with care due to the lack of penetration into the material.

14. Section 5.5, "Testing and Inspection," of the WP states, "For the testing program, one fuel assembly would be removed from the core at the end of the fourth year of operation, and then again after the 12th and/or 14th, 16th, and 18th year of operation."

    It is not clear to the NRC staff how an early phenomenon could be detected if it is not seen in periodic 4th-year removal but possibly occurs between 4th-year and 12th-year removal, and not detected by the core monitoring system.

SECTION 3

GENERAL OBERVATIONS

1. ARC indicated that BISON has been used as the primary modeling and simulation tool to assess the performance of U-10Zr fuel system. Although BISON has been validated for light-water reactor fuel, its history with metallic fuel is brief. It has not been submitted to the NRC staff for review and approval, so its use to support licensing actions would involve additional evaluation efforts. Future reviews of ARC licensing submittals would likely focus on the ability to accurately model the design, quantification of uncertainties associated with the fuel system, and the material properties and behavioral models that BISON incorporates. Additionally, the NRC staff would evaluate BISON's ability to model ARC's fuel design with specific geometrical and operational parameters that fall outside the previous operational experience (from EBR-II and FFTF), such as higher cladding exposure and longer residence time at elevated temperatures.

2. The use of consistent terminology throughout the WP would ensure technical accuracy. For example, the distinction between "irradiation-induced" and "irradiation-enhanced" is crucial, as these terms imply two different phenomena. In the former, irradiation is the primary driver, while in the latter, another factor (such as temperature) is the primary driver, with irradiation accelerating or magnifying the effects. When the terms are used interchangeably for both fuel and cladding materials, it creates confusion. For the review of this WP, the NRC staff assumed the following creep terminology:

    a. Irradiation-Induced Creep: The total creep of the material under irradiation. While this includes thermal creep, creep caused by irradiation is more dominant. Therefore, this term is more appropriate for the fuel zone.
    b. Irradiation-Enhanced Creep: This term refers to the total creep of the material due to

irradiation at elevated temperatures. In this case, thermal creep is dominant, but irradiation accompanies. Therefore, this term is more appropriate for materials other than the fuel.

c. Irradiation Creep: Creep component due to irradiation only.
d. Thermal Creep: Creep component due to temperature only.

3. The NRC staff recognizes and appreciates the efforts placed in the report; however, the following example editorial observations or inconsistencies are noted:

   a. Page 48 states, "Figure 3.1 shows data for HT9…," but Figure 3.1, "Fuel Qualification Program Elements" on page 44 shows the proposed roadmap instead. It appears page 48 should have referenced Figure 3.2, "Summary of HT9 Pressurized Tube Failure Strain Data" on Page 49.
   b. Page 55 states, "Figure 3.2 shows the greatest effect noted in these studies…," but it appears the correct cross-reference should be Figure 3.3, "Comparison of the unirradiated and irradiated Charpy curves for one third size specimen of 12Cr-1MoVW (HT9) and 9Cr-2WVTa steels in FFTF at 365ºC" on page 56.
   c. Page 106 states, "Results from BISON calculation of the end-of-life design criteria characteristics for ARC-100 peak fuel pin are shown in table 4," but it should have referred to table 5-6 "End-of-life design criteria characteristics for ARC-100 peak fuel pin," instead.
   d. Bibliography reference 88 is listed as BNL-NUREG-2102. The NRC staff ascertained reference 88 may be referring to BNL-NUREG-2012 instead, a memo dated September 1, 2016, that transmits "Assessment of the Adequacy of Metallic Fuel Qualification to Support the Licensing of Small Modular Sodium-Cooled Fast Spectrum Reactors" (4-S, ARC-100, PRISM) (ML16248A002).
   e. Bibliography reference 12 points to an invalid link.
   f. Figure 3.3 includes a caption citing the figure source; similar captions should be included on other figures, graphs, or images reproduced from another source (e.g., Figure 3.2 on page 49).
   g. Some figures, graphs, and tables were not legible (e.g., Figure 3.2 on page 49; Figure 4.1 "**[[** **]]**" on page 58; appendix A, "**[[** **]]**" on pages 146-148).

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