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To:	Meg Audrain; Jordan Hoellman
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Subject:	[External_Sender] FW: input / RE: separate ASME comments / RE: [EXTERNAL] Fwd: For Review and Comment: Material Compatibility ...
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Attachments:	image001.png Comments on NRC interim staff guidance.docx

Hi Meg and Jordan

We were not able to submit the comments to the Fed Reg on time. I thought I would send you the comments anyway, just for your information.

No action is requested.

Thanks

-Sam

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Subject: Comments on "Draft Interim Staff Guidance Material Compatibility for Non-Light Water Reactors, DANU-ISG-2023-01," Reference Docket ID NRC–2022–0215

Background

The draft Interim Staff Guidance (ISG) identifies areas of staff review that could be necessary for a submittal seeking to use materials allowed under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for the Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors" (Section III-5) (ASME, 2017). Section III-5 specifies the mechanical properties and allowable stresses to be used for design of components in high-temperature reactors (HTRs) but does not provide methods to evaluate deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities.

General Comment 1.

We strongly support the emphasis on performance monitoring and surveillance specimens/testing to address materials degradations during reactor operations.

We note that the need for performance monitoring and surveillance programs, particularly for very long design lifetimes, e.g., 500,000 hour, has been reinforced in Division 5 recently. A General Note has been added to Table HBB-I-14.10E-1 on the stress rupture factors for 9Cr-1Mo-V weldment in the 2023 edition of Division 5 which states:

> *The values in this table are extrapolated from shorter term test data using an engineering model. For longer design lives, the designer should consider further strength reductions to account for potential in-service material degradation, per HBB-2160(a). In addition, enhanced material surveillance programs and/or heightened in-service inspection per the rules of ASME Section XI may be warranted.*

General Comment 2.

Since some Advanced Non-Light Water Reactors (ANLWRs) may be used in whole or in part for the generation of nuclear process heat that will be used in an associated facility (e.g., hydrogen generation, ammonia production, petrochemical refining, etc.), it will be very important to provide guidance as to where the nuclear island stops and the non-nuclear facility begins with regard to safety standards and design margins for the structures, systems, and components (SSCs). This is particularly important for secondary or tertiary heat transfer loops. Also, the potential for adverse feedback between the nuclear and non-nuclear portions of a site--going either way--must be considered.

General Comment 3.

How will the safety significance and potential consequences of SSC malfunctions or failures be assessed to set required margins and assess design adequacy? Considering that one goal of ANLWRs is to produce power more economically, vendors will likely consider the use of reduced margins or commercial design codes where possible/appropriate. Augmented staff guidance to evaluate the adequacy of these approaches should be provided.

Comment 1.

It is agreed that appropriate mitigation strategies, performance monitoring, and surveillance programs should be considered to address the effects of thermal aging on design properties (ISG p8). Though, it is noted that thermal aging effects on yield and ultimate strength are specifically addressed in Division 5 and mandatory factors are provided.

Comment 2.

While components fabricated with advanced manufacturing technology (AMT) are addressed in the ISG (pp 9-10), there does not appear to be guidance regarding assuring that their high temperature properties are adequately defined. While this is an emerging field, limited studies have shown that while some AMT materials may have comparable room temperature or even short-term elevated temperature properties, long-term creep, fatigue, and creep-fatigue properties may be significantly reduced from wrought material values. Highlighting this issue and providing guidance or references related to it would be valuable. The ASME Section III Task Group on Division 5 AM Components would be a good source for up-to-date information on this subject.

Comment 3.

While wear and fretting are mentioned in the ISG (pg 9), it is important to note that tribology is significantly affected by particular coolants. Limited results have shown issues related to self-welding of SSCs in helium coolants. The virtual elimination of external oxide layers on metallic components in fluoride salt coolants practically ensures different tribological behavior in such media. Staff need to be given guidance to assess tribology in specific reactor environments.

Comment 4.

Guidance on ensuring that irradiation effects are adequately addressed for ANLWRs is described in multiple places in the ISG. However, given that the high doses likely to be reached in some fast reactor components can significantly exceed the existing data base on irradiation effects, it could be valuable to provide additional guidance regarding the potential value and limitations for using high-dose, ion-beam irradiations to further assess property changes at these high doses. The recent work funded as part of DOE's Integrated Research Program and lead by the University of Michigan on high-dose ion-irradiation effects probably provides the most comprehensive information on this subject currently available.