

Enclosure 2

**KP-FHR Metallic Material Qualification Topical Report
(Non-Proprietary)**

CONTENTS

<u>Section</u>	<u>Description</u>
A	Kairos Power LLC – Final Safety Evaluation for Topical Report Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (Revision 4), dated March 13, 2023.
B	Kairos Power Topical Report: “Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” KP-TR-013-NP-A, Revision 4.

Section A



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

March 13, 2023

Mr. Peter Hastings
Vice President, Regulatory Affairs
and Quality
Kairos Power LLC
707 W Tower Ave.
Alameda, CA 94501

**SUBJECT: KAIROS POWER LLC – FINAL SAFETY EVALUATION FOR TOPICAL
REPORT “METALLIC MATERIAL QUALIFICATION FOR THE KAIROS POWER
FLUORIDE SALT-COOLED HIGH-TEMPERATURE REACTOR” (REVISION 4)
(EPID NO. L-2021-TOP-0022/CAC NO. 000431)**

Dear Mr. Hastings:

This letter provides the final safety evaluation (SE) for the Kairos Power LLC (Kairos) topical report (TR) “Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor,” Revision 4. By letter dated June 30, 2020, Kairos submitted Revision 0 of the TR for the U.S. Nuclear Regulatory Commission (NRC) staff review (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20182A799). On November 25, 2020, the NRC staff provided its review questions to Kairos (ML20332A076). The review questions were discussed during a closed meeting held between the NRC staff and Kairos on January 13, 2021. By letter dated June 30, 2021, Kairos submitted Revision 1 of the TR (ML21181A385) to address the NRC staff review questions. By letter dated April 26, 2022, Kairos submitted Revision 2 of the TR (ML22116A246) to address questions discussed during a closed meeting held between the NRC staff and Kairos on February 2, 2022, and February 14, 2022. By letter dated August 19, 2022, Kairos submitted Revision 3 (ML22231B221) to address technical items discussed with the NRC staff during closed meetings held on July 18, 2022, and August 10, 2022. By letter dated September 20, 2022, Kairos submitted Revision 4 (ML22263A456) to address technical items discussed with the NRC staff during a closed meeting held on September 12, 2022.

The NRC staff’s final SE for “Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor,” Revision 4, is enclosed. The NRC staff provided Kairos a draft of the SE for the purpose of identifying proprietary information on December 19, 2022 (ML22355A307). On December 23, 2022, Kairos confirmed that the proprietary information in the draft SE was appropriately marked (ML23058A225).

Enclosure 1 to this letter contains Proprietary and Export Controlled Information. When separated from Enclosure 1, this letter is DECONTROLLED.

P. Hastings

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The Advisory Committee on Reactor Safeguards (ACRS) was briefed on this TR on January 12, 2023, and February 1, 2023. The ACRS provided its recommendations for the publication of this SE in a letter dated February 15, 2023 (ML23037A951). The enclosed SE is final, and a redacted version will be made publicly available.

The NRC staff requests that Kairos publish an accepted version of this TR within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed SE after the title page. The accepted version shall include an "-A" (designating accepted) following the TR identification number.

If you have any questions, please contact Richard Rivera at Richard.Rivera@nrc.gov.

Sincerely,



Signed by Jessup, William
on 03/13/23

William Jessup, Chief
Advanced Reactor Licensing Branch 1
Division of Advanced Reactors and Non-Power
Production and Utilization Facilities
Office of Nuclear Reactor Regulation

Project No.: 99902069 and Docket No.: 05007513

Enclosures:

1. Proprietary (Non-Public) Safety Evaluation
2. Redacted (Public) Safety Evaluation

cc via ListServ: Distribution at Kairos Power FHR and Kairs Power Hermes

P. Hastings

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SUBJECT: KAIROS POWER LLC – FINAL SAFETY EVALUATION FOR TOPICAL
REPORT “METALLIC MATERIAL QUALIFICATION FOR THE KAIROS POWER
FLUORIDE SALT-COOLED HIGH-TEMPERATURE REACTOR” (REVISION 4)
(EPID NO. L-2021-TOP-0022/ CAC NO. 000431)
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NRR-106

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UNITED STATES
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March 13, 2023

**FINAL SAFETY EVALUATION OF METALLIC MATERIAL QUALIFICATION FOR THE
KAIROS POWER FLUORIDE SALT-COOLED HIGH-TEMPERATURE REACTOR
(KP-TR-013) KAIROS POWER, LLC EPID NO. 000431 / 99902069 / L-2020-TOP-0050**

1.0 SPONSOR INFORMATION

Sponsor: Kairos Power, LLC (Kairos)

Address: 707 West Tower Ave.
Alameda, CA 94501

Project No.: 99902069 (Construction Permit Application Docket No. 05007513)

2.0 SUBMITTAL, CORRESPONDENCE, AND CONTRIBUTORS

2.1. Submittal Information

Revision 0	June 30, 2020	ML20182A799	KP-TR-014, Revision 0
Revision 1	June 30, 2021	ML21181A385	KP-TR-014, Revision 1
Revision 2	April 2, 2022	ML22116A246	KP-TR-014, Revision 2
Revision 3	August 19, 2022	ML22231B221	KP-TR-014, Revision 3
Revision 4	September 20, 2022	ML22263A456	KP-TR-014, Revision 4

*Agencywide Documents Access and Management System (ADAMS) Accession No.

2.2. NRC Correspondence and Communications

Communication Type	Date	ADAMS Accession No.
Acceptance Review(s):	September 3, 2020	ML20224A172
Closed Meeting Notices:	December 6, 2021	ML21336A400
	February 3, 2022	ML22032A336
	February 14, 2022	ML22032A336
	July 18, 2022	ML22196A385
	August 10, 2022	ML22214A131
	September 12, 2022	ML22244A250

*ADAMS Accession No.

2.3. Principal Contributor(s)

- John Honcharik, NRR/DNRL/NPHP
- Alexander Chereskin, NRR/DANU/UTB2
- Richard Rivera, NRR/DANU/UAL1

3.0 BRIEF DESCRIPTION OF REQUEST AND BACKGROUND

Kairos Power, LLC (Kairos, the sponsor) is requesting Nuclear Regulatory Commission (NRC) staff review and approval of topical report (TR) KP-TR-013, “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor,” Revision 2, dated April 2022. The TR could apply to reactors using the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR) designs¹ and could be used to support future licensing actions under Title 10 of *Code of Federal Regulations* (10 CFR) Parts 50 or 52. The TR includes the qualification plan for metallic structural materials used in Flibe-wetted areas for safety-related high temperature components of the KP-FHR power and non-power (test) reactors. Kairos also requested NRC approval of the planned material testing and analyses to address the materials reliability and compatibility in the environment of the KP-FHR designs. The results of these planned tests and analyses will be provided in a future license application that references this TR, along with a detailed description of the design, inspection, and surveillance programs for the KP-FHR designs.

The documents located at the ADAMS Accession number(s) identified in Section 2 of this SE have additional details on the submittal.

4.0 EVALUATION CRITERIA

4.1 Regulatory Requirements

The information Kairos will gather through their metallic material qualification program will satisfy, in part, 10 CFR 50.10, 10 CFR 50.34, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, 10 CFR 52.157, which describe the requirements for the content of applications of limited work authorizations, construction permits, operating licenses, design certifications, combined licenses, standard design approvals, and manufacturing licenses, respectively.

4.2 Principal Design Criteria for the KP-FHR, Approved by the NRC Staff

The topical report KP-TR-003-P-A, “Principal Design Criteria (PDC) for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” Revision 1, dated May 2020, provides PDCs for the KP-FHR design that were reviewed and approved by the NRC staff. The PDCs below are applicable to qualification of metallic components for the KP-FHR designs.

KP PDC 14, “Reactor coolant boundary,” which requires safety significant elements of the reactor coolant boundary to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The continued performance of high temperature structural materials and the associated corrosion within the coolant relate to PDC 14.

¹ When the term “KP-FHR designs” is referenced in this safety evaluation (SE), it applies to both the power reactor and non-power test reactor, unless otherwise specified.

KP PDC 31, "Fracture prevention of reactor coolant boundary," which requires, in part, the reactor coolant boundary to behave in a nonbrittle manner and to minimize the probability of rapidly propagating failure of the reactor coolant boundary, accounting for effects of coolant composition on material properties. The design reflects consideration of service temperatures, service degradation of material properties, creep, fatigue, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions, and the uncertainties in determining: (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.

4.3 Codes, Standards, and Guidance Documents

Applicable Codes and Standards:

The NRC staff also considered the following codes and standards and guidance documents during the course of its review:

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III Division 5, "Rules for Construction of Nuclear Power Plant Components, High Temperature Reactors," 2017 Edition.

Guidance Documents:

NUREG-2245, "Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, "High Temperature Reactors"" dated January 2023 (ADAMS Accession No. ML23030B636)

Regulatory Guide (RG) 1.87, "Acceptability of ASME Code Section III, Division 5, High Temperature Reactors," Revision 2, dated January 2023 (ADAMS Accession No. ML22101A263)

5.0 STAFF EVALUATION

5.1 Staff Evaluation Discussion

Kairos submitted this TR regarding the development of its safety-related reactor coolant boundary to support future licensing actions for reactors using the KP-FHR designs under 10 CFR Parts 50 or 52, including KP-FHR power reactors and non-power test reactors. The TR describes the qualification and testing methodology to be used for the metallic structural materials in safety-related components exposed to the high temperature reactor coolant salt (known as Flibe) environment of the KP-FHR designs. The Flibe properties are provided in the Kairos Power TR, "Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor," Revision 1 (ML20016A486), which was approved in an NRC staff SE dated July 16, 2020 (ML20139A224).

As stated in Section 5.1 of the TR, the sponsor requested NRC staff to review and approve the qualification requirements for environmental effects of Flibe on the metallic structural materials provided in Section 4 of the TR, which the applicant has proposed will partially satisfy PDC 14 and PDC 31. The qualification requirements provided in Section 4 of the TR are for environmental effects of Flibe on the metallic structural materials, which are in addition to the qualification requirements for mechanical properties of 316H austenitic stainless steel and ER16-8-2 stainless steel weld filler metal required by ASME Code, Section III, Division 5. The applicant stated that a description of how the remaining portions of these PDC are satisfied will be provided in safety analysis reports submitted with license applications for the KP-FHR designs. The applicant stated that these material qualification test results will be used as a basis in future licensing actions to address potential materials reliability and environmental compatibility issues via design, operation, and inspection.

The results of the planned tests and analyses, along with a description of the design, operation, inspection, and surveillance programs to manage the materials performance, will be provided in future license applications. The remainder of the TR was not evaluated by the NRC staff and was only reviewed as technical background and to identify any potential impacts on the portions of the TR for which Kairos requests approval. Therefore, KP-FHR designs referencing this TR may only use this TR for purposes related to the information on 316H and ER16-8-2 material found in Section 4 of the TR, subject to the specific Limitations and Conditions found in Section 6.0 of the NRC staff SE below. All other information related to 316H and ER16-8-2 material will be evaluated in separate documents and licensing actions (see Limitation and Condition 1).

As stated in Sections 1.1.3.2 and 5.1 of the TR, the reactor vessel is [REDACTED] safety-related component exposed to Flibe that is required to keep the fuel covered in Flibe during all normal operations and postulated events. The environmental effects qualification testing in this TR was based on the environment that the reactor vessel would experience. Therefore, the environmental effects qualification testing for the KP-FHR designs in this TR can only be used for other components with environments that are bounded by the environment the reactor vessel would experience and referenced in this TR. For example, other components that would have Flibe on one side of the metallic material and another salt on the other side of the metallic material, or would be exposed to higher irradiation levels than those specified in the TR, or be subject to conditions otherwise not addressed in the TR would not be bounded by this TR (see Limitation and Condition 2.)

The metallic structural materials proposed for the KP-FHR designs are 316H austenitic stainless steel and the associated ER16-8-2 stainless steel weld filler metal which are qualified for use in ASME Code, Section III, Division 5, for high temperature reactors. The NRC staff notes that 316H and ER16-8-2 are materials that can be used in high temperature reactors since these materials are qualified materials listed in ASME Code, Section III, Division 5. ASME Code, Section III, Division 5, provides minimum quality requirements for the materials to ensure the use of the materials will result in an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture, which partially satisfies PDC 14 and PDC 31. The NRC staff has endorsed the use of ASME Code, Section III, Division 5 as per NUREG-2245, "Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, "High Temperature Reactors", (ML23030B636) and Regulatory Guide 1.87, "Acceptability of ASME Section III, Division 5, High Temperature Reactors," (ML22101A263).

However, ASME Code, Section III, Division 5, specifies ER16-8-2 weld filler metal as a qualified material for use up to 650°C while 316H is qualified for use up to 816°C. Currently, testing is being performed in accordance with the ASME Code to extend the qualification of ER16-8-2 up to [REDACTED] which includes the extension of the material's stress rupture factors to the higher temperature. This testing is addressed below in further detail.

Although ASME Code, Section III, Division 5, contains stress rupture values up to [REDACTED], the staff endorsement in RG 1.87 imposes a limitation to not endorse all the stress rupture values found in Table HBB-I-14.6B, "Expected Minimum Stress-to-Rupture Values, 1,000 psi (MPa), Type 316 SS." The NRC staff limitation provides tables to show acceptable use of the stress rupture data based on the amount of time at a specified temperature [REDACTED]. However, because Kairos stated that [REDACTED]

[REDACTED]. The NRC staff finds this to be acceptable because the time at the specified temperature, for both normal operations and postulated accidents, falls within the NRC staff-endorsed ranges found in Table 2 of Regulatory Guide 1.87 for 316H. If the time and temperature for both normal operations and postulated accident conditions change for the KP-FHR designs, they must still be bounded by the NRC staff-endorsed ranges found in Table 2 of Regulatory Guide 1.87 for 316H, or an adequate justification must be provided for NRC staff review and approval as to why the values outside of the endorsed ranges are acceptable. (see Limitation and Condition 3.)

Since ER16-8-2 is not currently qualified to the higher temperature necessary to support accident scenarios of the KP-FHR designs, the NRC staff imposes a condition that ER16-8-2 must be qualified to a temperature of [REDACTED] in accordance with the requirements of ASME Code, Section III, Division 5, that bounds the postulated accident conditions. The qualification must also be approved by the NRC staff (see Limitation and Condition 4).

5.1.1 Design of the KP-FHR

Section 1.1 of the TR provides an overview of the key design features of the KP-FHR designs. The applicant stated that these features are not expected to change during the development of the KP-FHR designs. The applicant also stated that these features provide the basis for the safety review of the TR and that if fundamental changes occur to the key design features, or new or revised regulations are issued, these changes would be reconciled and addressed in

future submittals. Because the TR is requesting approval of certain characteristics of the reactor coolant boundary without the full scope of knowledge of detailed system specifications, there may be instances where the design features, as outlined in the TR, change between submittal of this TR and a future licensing action. Accordingly, the NRC staff added a condition and limitation to the TR contingent on the design features provided in Section 1 of the TR (see Limitation and Condition 5).

5.1.2 Environment to be Tested

The environments for both the non-power (test) reactor and the commercial power reactor are specified in Table 1 of the TR and are similar except that the non-power reactor lifetime is 5 years, as opposed to [REDACTED] for the commercial power reactor. The operating environment parameters for the KP-FHR designs concerning environmental degradation include the following:

- Flibe salt temperatures of 550°C-650°C
- An intermediate salt coolant loop for the commercial reactor
- A Primary Heat Transport System that rejects heat to the air in lieu of an intermediate coolant loop for the non-power test reactor
- Non-power test reactor lifetime of 5 years (1 year commissioning and 4 years operation) and commercial power reactor lifetime of [REDACTED]
- “Near-atmospheric” primary coolant pressures
- End of life irradiation of less than 0.1 displacement per atoms (dpa)

These are key operating environment parameters necessary to develop the qualification testing of 316H and ER16-8-2 for specific environmental degradation mechanisms. Therefore, the NRC staff is imposing a limitation and condition that KP-FHR designs referencing this TR must have the key operating environment parameters described above and, if changed, could necessitate the modification of, or addition to, the testing program. (see Limitation and Condition 6).

Table 11 of the TR provides the specific degradation mechanics of 316H and ER16-8-2 for the operating environment in the KP-FHR designs with the associated testing to determine the effects the operating environment has on these materials. The NRC staff finds that environmental effects testing at the normal operating temperatures to validate the degradation of 316H and ER16-8-2 material is acceptable since it duplicates the environment the material would experience during operation. Also, the additional testing using higher test temperatures [REDACTED] will allow the applicant to develop environmental degradation rates that may be experienced during postulated accident scenarios [REDACTED]

[REDACTED] The NRC staff finds that the test temperature of [REDACTED] can be used to quantify any increase in degradation of the material during a postulated accident scenario with a maximum temperature of [REDACTED] for use in future licensing applications, in part, to satisfy PDC 31. However, if the postulated accident scenarios [REDACTED], the test temperature and

time for the associated material testing in Section 4 of the TR should be increased to [REDACTED] (see Limitation and Condition 7).

As stated in Section 4.2.3 of the TR, most of the testing will be conducted in “Nominal Flibe”, i.e., Flibe which has been purified to minimize water and other oxidizing contaminants but not with excess beryllium metal to invoke redox control (i.e., Redox Controlled Flibe). The NRC staff finds that material testing in Nominal Flibe will bound the materials (316H and ER16-8-2) in Redox Controlled Flibe because Nominal Flibe has a higher oxidizing potential leading to increased degradation rates than in Redox Controlled Flibe. Redox Controlled Flibe uses [REDACTED] which reduces the concentration of tellurium and the oxidizing potential in Nominal Flibe, thereby leading to potentially lower degradation rates in Redox Controlled Flibe. Therefore, the NRC staff finds that the material testing in Redox Controlled Flibe can be used as a sensitivity study to [REDACTED]

[REDACTED] that can be used in future license applications for the KP-FHR designs. However, the use of [REDACTED] in Flibe has the potential to form intermetallic phases in 316H and ER16-8-2 as noted in Reference 7. This potential effect is addressed in Section 5.1.3.3.2 of this SE with associated Limitation and Condition 11, to determine the effects of [REDACTED] on the mechanical properties of 316H and associated weld filler metal ER16-8-2.

Section 4.2.3.3 of the TR describes two potential accident scenarios for the commercial power reactor (i.e., intermediate salt ingress for [REDACTED]) and air ingress for [REDACTED] into the Flibe salt) that would produce a specific concentration of these impurities that could affect the safety-related components. [REDACTED]

[REDACTED] Therefore, Tables 12 and 13 of the TR, as described in Section 4.2.3.3, provide the proposed impurity testing for both salt and air that will cover accident scenarios postulated in the transient safety analyses, and originally defined in the materials Phenomena Identification and Ranking Table (PIRT) review. In addition, the ingress of air impurities is also accounted for and tested in combination with the intermediate salt from the intermediate loop for the power reactor. The NRC staff finds this approach acceptable for developing the effect on corrosion rates that both air and the intermediate salt may have on 316H and ER16-8-2 because it will bound the accident conditions for the power reactor. The NRC staff also finds that performing corrosion testing of 316H and ER16-8-2 in Nominal Flibe with air (as an impurity) for up to [REDACTED] provides a reasonable method of developing corrosion rates in Nominal Flibe with impurities for the non-power test reactor. The NRC staff also notes that the details of the impurity testing (e.g., the concentration of contaminant) have not been determined, as stated in Table 13 of the TR. Therefore, the specific conditions of the impurities in Nominal Flibe, including contaminant chemistry, used in the impurity effects testing on 316H and ER16-8-2 shall bound the accident scenarios postulated in the transient analyses documented in the safety analysis reports for the KP-FHR designs (see Limitation and Condition 8).

5.1.3 Degradation Mechanisms

The TR provides the necessary material testing to determine the rate of degradation of 316H and ER16-8-2 in the environment of the KP-FHR designs using Flibe. The test results will be used to confirm that safety-related reactor coolant boundary material under operating and

postulated accident conditions have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture, which partially satisfies the criteria in PDC 14 and 31. The material testing of 316H and ER16-8-2 in Flibe will be conducted for the following degradation mechanisms:

- Corrosion (including general corrosion, crevice corrosion, thermal aging, erosion/wear and cold leg occlusion)
- Environmentally assisted cracking (including stress corrosion cracking, environmental creep, and corrosion fatigue)
- Effects on metallurgical properties (including stress relaxation cracking, phase formation embrittlement, and thermal cycling)
- Irradiation effects (including irradiation-affected corrosion, irradiation-assisted stress corrosion cracking, and irradiation-induced embrittlement)

References 11 and 12 to the SE describe various degradation mechanisms, whether they occur in molten salt environments, and where additional information may be needed. These references identify corrosion, environmentally assisted cracking, and the effects of irradiation on materials as subjects where knowledge gaps may exist and require additional study. Reference 11 identifies that more data is needed for corrosion in molten salt including the effects of impurities and redox control on corrosion rates, and that there is a knowledge gap for environmentally assisted cracking in molten salts. This reference also notes that irradiation may affect degradation of material in molten salts, but that little data is currently available. Reference 12 identifies the potential for formation of intermetallic phases and the corresponding reduction in material strength. Therefore, the NRC staff finds that the above environmental degradation mechanisms are pertinent to 316H and ER16-8-2 in Flibe and are consistent with information needs identified in currently available research data and testing described above and in the TR.

Therefore, the NRC staff finds that the TR can be used in future licensing actions for the above degradation mechanisms described in Section 4 of the TR for the KP-FHR designs to partially satisfy PDCs 14 and 31, subject to the Limitations and Conditions found in Section 6.0 of the NRC staff's SE. The specific evaluation for the testing of each degradation mechanism is provided below. The NRC staff notes that additional information and research on different degradation mechanisms may become available in the future. These different degradation mechanisms would require additional testing and would be evaluated in future licensing actions.

5.1.3.1 Corrosion

Section 4.2.3 of the TR provides an overview of the proposed corrosion testing that will be used to develop quantitative corrosion models for 316H stainless steel in a Flibe environment. The NRC staff did not make a finding with regards to the overview of the proposed corrosion testing in Section 4.2.3.

5.1.3.1.1 Corrosion Test Systems

Section 4.2.3.1 of the TR describes the systems that were developed to perform corrosion tests. Kairos stated that the [REDACTED]

degradation mechanisms. This includes tests to determine the effects of temperature, microstructure, salt composition, geometry, erosion-corrosion, thermal aging, graphite contact, and difference in solubility of corrosion products on the corrosion rate of 316H and ER16-8-2.

The NRC staff evaluated the planned corrosion testing for the KP-FHR designs that is summarized in Section 4.2.3.3, and Tables 12 and 13 of the TR. The staff also evaluated the proposed method to determine corrosion kinetics and the steady state corrosion rate, which are described in Section 4.2.3.3 and Appendix C of the TR. The NRC staff finds the proposed corrosion testing acceptable because these tests will determine the impact of temperature, microstructure, salt composition, geometry, erosion-corrosion, thermal aging, presence of graphite, redox control, and difference in corrosion product solubility (i.e., cold leg occlusion) on the corrosion rates and corrosion kinetics of 316H and ER16-8-2. In addition, these tests are acceptable because they are consistent with the expected corrosion mechanisms for 316H and ER16-8-2 in a molten salt environment (Raiman 2021) and a portion of the tests will be conducted with flowing Flibe, which is necessary to simulate the flowing salt in a reactor.

The NRC staff finds the tests to determine the effect of temperature on corrosion rates acceptable because corrosion is evaluated over a range of temperatures consistent with the operating temperatures of the KP-FHR designs including bounding postulated accident conditions which satisfies PDCs 14 and 31, in part. In addition, the NRC staff finds the test durations will provide sufficient data to determine corrosion kinetics.

The NRC staff also finds the tests to evaluate the microstructural effects on corrosion rates acceptable because, as described in Table 12 of the TR, these include tests to examine effects of [REDACTED] which are known to increase corrosion rates.

The NRC staff finds that the tests using both the Nominal Flibe composition, as well as those tests with a reducing agent added, are acceptable because these tests will determine the effects of the Flibe composition, including how oxidizing contaminants, as well as redox control, affect the corrosion rate. These tests will provide data necessary to determine design margins for corrosion, allowable levels of impurities in the salt, and the potential benefit from adding a redox control agent. An applicant referencing this TR must demonstrate that the salt compositions (with reducing agent additions and impurities from postulated accident scenarios) tested in this program bound any potential salt compositions for the KP-FHR designs (see Limitation and Condition 10).

With regard to occluded geometry effects on corrosion rates, the NRC staff finds the proposed tests acceptable because these tests will determine whether crevice corrosion is a concern for 316H and ER16-8-2 in Flibe, and the potential effect on the corrosion rate.

The NRC staff finds the proposed tests to determine the effect of erosion-corrosion acceptable because the tests will utilize graphite particulate to determine the effect of these particles on corrosion rates, as well as [REDACTED]. This is necessary because the KP-FHR designs will utilize graphite pebbles, as well as a graphite reflector, which will introduce graphite dust into the Flibe. Additionally, it is appropriate to determine the effect of graphite on corrosion because the presence of graphite can accelerate corrosion of 316H and ER16-8-2 when in the same system as the fluoride salt (Flibe).

The NRC staff finds the tests to determine the impact of cold leg occlusion acceptable because the proposed tests have a temperature differential between the hot and cold legs consistent with

the KP-FHR designs. This temperature differential is necessary because corrosion products are more soluble in the hot leg and will precipitate out in the cold leg. This creates a concentration gradient that will accelerate corrosion as a function of the temperature differential between the hot and cold legs and will be simulated in the tests.

5.1.3.1.4 Corrosion Modeling

Section 4.2.3.3 of the TR stated that testing will be used to analyze the depth of Chromium loss over time to establish the corrosion kinetics and to determine the steady state corrosion rate. The depth of Cr loss and other metallurgical changes will be analyzed using electron microscopy. Appendix C, "Data Analysis", of the TR stated that this will allow for more sensitive measurements than analyzing the weight change of the test coupons. This is because measuring weight change can be complicated due to factors such as carbon pickup or difficulties in removing dried salt from the coupons. Electron microscopy will instead allow Kairos to analyze coupon cross sections to assess corrosion and other compositional changes.

Appendix C of the TR also stated that baseline corrosion models will be developed and separate effects tests will assess key variables that may impact corrosion rates. Kairos also stated that it will perform statistical analysis on the data and will utilize prediction bands to ensure appropriate and conservative extrapolation to the KP-FHR operational times and temperatures. For test data of certain degradation mechanisms (e.g., stress corrosion cracking) that may not be amenable to statistical analysis, Kairos stated that testing will be performed to detect if the phenomenon occurs, and whether variables that impact stress corrosion cracking can be quantified in order to perform a statistical analysis on the data. In scenarios such as this, Kairos stated that other practices (e.g., periodic inspections) may be used to address such phenomena, if the test data is not amenable to performing a statistical analysis.

The NRC staff evaluated the proposed corrosion modelling by Kairos in order to determine if the proposed qualification program for the KP-FHR designs will be adequate to determine performance of 316H and ER16-8-2 when exposed to the molten Flibe reactor coolant. The staff finds it acceptable to model corrosion behavior as a function of Cr loss from the 316H and ER16-8-2 because Cr is the alloying element in 316H that is most thermodynamically favored to corrode (i.e., least noble) and therefore will likely corrode prior to other elements of 316H and ER16-8-2 (DeVan, 1962, Raiman 2021). The staff also finds it acceptable to analyze the corrosion data as described in Appendix C because statistical analysis of the data will provide reasonable assurance that significant contributors to corrosion can be identified and that uncertainties resulting from the test data can be conservatively incorporated into corrosion predictions. Additionally, the staff finds use of electron microscopy acceptable because this will allow Kairos to assess the depth of Cr loss as well as other compositional changes in the material to mitigate complicating factors from the corrosion tests such as carbon pickup or difficulty removing dried salt from the material. This will provide data that can be corroborated against the observations from the electron microscopy. Use of electron microscopy is also acceptable because, as stated in Section 4.2.3.3 of the TR, weight change for each corrosion coupon will also be measured. The staff finds it acceptable to perform separate effects testing, in addition to baseline corrosion testing, because it will allow different variables to be assessed for their impacts on the corrosion rate. The staff finds it acceptable to perform some tests primarily to detect whether a specific phenomenon occurs, if the test data of a degradation mechanism is not amenable to statistical analysis, because after assessing whether a phenomenon occurs, it can be quantified and mitigated via multiple measures (e.g., inspections).

5.1.3.1.1.5 Effects of Operating Conditions on Corrosion

Section 4.2.3.4 of the TR describes how [REDACTED]

]].

The NRC staff finds it acceptable to [REDACTED]

]. The NRC staff will review the considerations discussed above to assess corrosion performance in a future submittal to determine whether the overall approach to assess and manage corrosion performance is acceptable.

5.1.3.2 Environmentally Assisted Cracking

5.1.3.2.1 Stress Corrosion Cracking and Corrosion Fatigue

Section 4.2.4 and Tables 14 and 15 of the TR provides the proposed material testing that will be performed to evaluate how the operating environment of the KP-FHR designs using Flibe affects the corrosion fatigue and stress corrosion cracking rates of 316H and ER16-8-2. Currently, there is little mechanical testing in molten salts due to the difficulty of conducting in-situ mechanical testing in highly reducing molten salt. There is also limited data of environmentally assisted cracking in stainless steels and nickel-based alloys in molten salts. Therefore, in-situ mechanical testing systems will be used to conduct slow strain rate testing (SSRT) for corrosion fatigue and stress corrosion cracking. Section 4.2.4.1 of the TR states that the SSRT tests will be conducted at temperatures [REDACTED] at various strain rates as described in Table 14 of the TR. The SSRT testing will be conducted in Nominal Flibe and Redox Controlled Flibe to assess if 316H, ER16-8-2, and the [REDACTED] of 316H are susceptible to environmentally assisted cracking in Flibe. Section 4.2.4.1 of the TR states that the SSRT testing will be conducted in accordance with American Society for Testing and Materials (ASTM) ASTM G129-00, "Standard Practice for Slow Strain Rate Testing to Evaluate the Susceptibility of Metallic Materials to Environmentally Assisted Cracking," 2000 Edition.

In addition to the SSRT testing, Section 4.2.4.2 of the TR states that fracture mechanics-based testing of pre-cracked compact tension type samples will be used to evaluate fatigue crack growth rates, stress corrosion cracking rates, and the fracture behavior of 316H and ER16-8-2 in the Flibe environment at temperatures [REDACTED] as described in Table 15 of the TR. The test samples will consist of [REDACTED] with a sharp flaw (i.e., the corrosion fatigue crack) placed in the appropriate locations (i.e., weld metal, [REDACTED] and base metal) and allows the crack to grow in the area to be tested. [REDACTED]

[REDACTED]

] These SSRT tests will include both a corrosion fatigue portion (high variable stress intensity factor to initiate and grow fatigue crack which will be used for subsequent stress corrosion testing) and stress corrosion cracking portion (a constant stress intensity factor to initiate stress corrosion under aggressive testing conditions and then transition to conditions that are representative of the KP-FHR designs).

The NRC staff finds that the samples to be tested are acceptable because they are representative of the material and weldments to be used in the KP-FHR designs. In particular, the use of [REDACTED]

[REDACTED]

] The NRC staff also finds that performing the testing for fatigue and stress corrosion cracking in air [REDACTED] and then comparing it to testing performed in both Nominal Flibe and Redox Controlled Flibe acceptable in simulating and determining the crack growth rates for fatigue and stress corrosion cracking relative to the environment in the KP-FHR designs.

[REDACTED]

] The NRC staff also finds that SSRT and fracture mechanics-based testing (ASTM G129-00) of pre-cracked compact tension samples acceptable because these test methods are well-established, are an accepted methodology, and have been effective in other environmentally assisted cracking testing used by licensees and the NRC staff.

5.1.3.2.2 Environmental Creep

Section 4.2.4.3 of the TR describes the material testing that will be performed to assess the susceptibility of 316H and ER16-8-2 to environmental creep in [REDACTED] as described in Table 16 of the TR. The NRC staff finds that performing creep testing in both Normal Flibe using welded base metal samples to include the base metal, weld metal and heat affected zone of the base metal acceptable since it simulates the material to be used and the environment of the KP-FHR designs. In addition, the NRC staff finds it acceptable that [REDACTED] are not required to be performed unless significant degradation is noted compared to creep tests performed in air, such as failure times outside of the 90% confidence interval from creep tests performed in air, or a change in fracture mode because the creep tests in air would bound the results in Nominal Flibe in the temperature range of the KP-FHR designs. The NRC staff notes that the creep tests in Nominal Flibe are to determine if the Flibe contributes additional degradation beyond those determine from the creep tests performed in air. If the testing determines Flibe has an additional effect on degradation, additional testing would be required to quantify any increase in degradation contributing to Flibe, and the test results would be reviewed by the NRC staff in future licensing submittals by the applicant.

5.1.3.3 Metallurgical Effects

5.1.3.3.1 Stress Relaxation Cracking

Section 4.2.5 of the TR states that stress relaxation cracking in 316H will be addressed by using test results conducted in air up to temperatures of [REDACTED] as discussed in Section 3.2 of the TR, and by conducting future analysis and design refinements of the KP-FHR designs, such as weld designs in Figure 23 of the TR, and specific weld processes and parameters to minimize stress relaxation cracking as detailed in Section 3.3.1 of the TR to reduce the triaxial stresses. Section 3.3.2 of the TR provides [REDACTED]

[REDACTED] [REDACTED]. Current data and experience show that ER16-8-2 weld metals are not susceptible to stress relaxation cracking, while the heat affected zone of 316H base metal with high triaxial stresses is susceptible to stress relaxation cracking. In addition, Table 10 of the TR provides the specific, [REDACTED]

[REDACTED] [REDACTED]. Section 4.1 of the TR states that the stress relaxation cracking testing is summarized in Table 11 of the TR.

The NRC staff finds testing in air acceptable because these test results would be valid for 316H in Flibe for the KP-FHR designs since triaxial stresses are the major contributor to stress relaxation cracking. In addition, the NRC staff finds that comparing the susceptibility of 316H to that of 347 as discussed in Section 3.3.1 of the TR would allow a determination of the bounding triaxial stresses that could cause stress relaxation cracking in 316H. The NRC staff also finds the stress relaxation testing for the KP-FHR commercial power reactor and the non-power test reactor in Table 10 of the TR acceptable because the [REDACTED] [REDACTED] will provide reasonable data to assess the bounding triaxial stresses that could cause stress relaxation cracking in 316H. The NRC staff notes that in Section 3.3.2 of the TR Kairos states that it's developing the capability to model weld residual stresses using [REDACTED] [REDACTED] to better assess and limit the magnitude of the weld residual stresses induced during welding of 316H for the KP-FHR designs. The results of these analysis can be used in future licensing actions to address stress relaxation cracking of 316H in the KP-FHR designs.

5.1.3.3.2 Phase Formation Embrittlement

Section 4.2.5 of the TR discusses how the qualification program addresses phase formation embrittlement, and degradation from thermal cycling or thermal gradients. Kairos states that phase formation embrittlement may occur when 316H and ER16-8-2 picks up an element during its exposure to Flibe and forms a deleterious second phase. To address this, Kairos proposed to [REDACTED]

[REDACTED] [REDACTED].

The NRC staff reviewed the proposed method to address phase formation embrittlement. The NRC staff finds it acceptable because Kairos will [REDACTED] [REDACTED] [REDACTED]. This testing will determine whether this degradation mechanism occurs for the KP-FHR designs. Additionally, this is subject to Limitations and Conditions 2 and

5 which states that the results of the qualification testing are only applicable to the KP-FHR designs that is bound by the test conditions. In this case, a design that utilizes [REDACTED] [REDACTED]]. Therefore, the NRC staff is imposing a limitation and condition that if intermetallic formation occurs, an applicant will need to perform testing to quantify the effects on the mechanical properties of 316H and associated weld filler metal ER16-8-2 (see Limitation and Condition 11).

5.1.3.3.3 Thermal Cycling/Stripping

Table 11 of the TR states that thermal cycling [REDACTED] [REDACTED]]. In addition, Section 4.2.5 of the TR states that degradation of 316H and ER16-8-2 by large thermal transients could lead to high stresses resulting in thermal fatigue degradation. Kairos will address the thermal cycling by conducting analysis to refine the design and operation of the KP-FHR designs to mitigate large thermal gradients. The NRC staff finds it acceptable that future analysis, in lieu of testing, will be used to mitigate thermal cycling because the thermal gradients will be minimized through the use of appropriate design and operating conditions of the KP-FHR (power and non-power test reactor), as informed by the analysis. However, since the design has not been finalized and no testing will be conducted as part of this material qualification program, the NRC staff is imposing a limitation and condition that an applicant implementing this TR will address thermal cycling/stripping in future licensing submittals by minimizing the thermal gradients via appropriate design and operating conditions of KP-FHR designs based on analysis (see Limitation and Condition 12).

5.1.3.4 Irradiation Effects

5.1.3.4.1 Irradiation-Induced Embrittlement

Section 4.2.6.1 of the TR states that existing data indicates that tensile properties and fracture toughness of austenitic stainless steels, when tested at high strain rates and temperatures from 550°C to 650°C, are relatively unaffected by irradiation levels <0.1 displacement per atoms (dpa) with a helium content of 10 atomic parts per million (appm) in current light water reactor environments. However, at low strain rates, data shows irradiation-induced embrittlement can affect material properties such as tensile strength and ductility and creep life due to the generation of helium. The applicant stated in Section 4.2.6.1 of the TR that existing data will be used to develop degradation factors, but that it will conduct irradiation tests on ER16-8-2, 316H, and the associated heat affected zone of 316H to quantify margins at irradiation levels for the non-power test reactor and the commercial power reactor which will be provided in future licensing actions. The NRC staff finds it acceptable to conduct testing for irradiation-induced embrittlement on ER16-8-2, 316H, and the associated heat affected zone of 316H, because the testing will be representative of the environment in the KP-FHR designs and this information will be submitted in future licensing actions. NRC staff is imposing a limitation and condition that the test environment shall bound the KP-FHR designs, including the expected irradiation damage (dpa) and helium content (see Limitation and Condition 13).

5.1.3.4.2 Irradiation-Affected Corrosion

Section 4.2.6.2 of the TR states that no immediate material testing of 316H and ER16-8-2 for irradiation effects on corrosion is proposed for the qualification of 316H and ER16-8-2 because the reactor vessel has a low irradiation dose level (<0.1 dpa) and existing data shows that

irradiation may increase general corrosion rates but decrease intergranular corrosion rates. However, the applicant will implement a materials surveillance system program for the non-power test reactor and (at least the first) commercial power reactor systems to monitor irradiation-affected corrosion. In addition, an inspection and monitoring program that will assess the wall thickness of the reactor vessel will also be implemented. The initial plans for these programs are provided in Appendix B of the TR. The applicant has not finalized plans for these programs and will provide the detailed programs in future licensing actions.

Since Appendix B of the TR is not a finalized program for assessing irradiation-affected corrosion, the NRC staff cannot provide a conclusion on the proposed initial planned programs. Notwithstanding, the NRC staff finds it acceptable to implement a materials surveillance program that will be submitted as part of future license applications for the non-power test reactor and the commercial power reactor because this program could provide sufficient information that can be used in determining any effects irradiation has on the corrosion rate of 316H and ER16-8-2 in the environment of the KP-FHR designs. However, the NRC staff notes that the materials surveillance program should not be limited to only the first commercial power reactor, because there is limited data on the effects of irradiation on corrosion rates in Flibe on 316H and ER16-8-2. Therefore, the materials surveillance program should apply to both the non-power test reactor and the commercial power reactors. In addition, the NRC staff finds it acceptable to use an inspection and monitoring program to assess any changes in the wall thickness of the reactor vessel because the program should be capable of detecting wall thinning that could prevent the reactor vessel from performing its safety function. Therefore, the NRC staff is imposing a limitation and condition that the materials surveillance program and the inspection and monitoring program will be submitted in future license applications for NRC staff review and approval to verify that these programs are sufficient to address irradiation-affected corrosion of the reactor vessel. (See Limitation and Condition 14.)

5.1.3.4.3 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Section 4.2.6.3 of the TR states that IASCC is not expected to be a degradation mechanism in the KP-FHR design due to the low irradiation level (<0.1 dpa) and that radiolysis of Flibe is not expected because of the rapid recombination of ions in the molten Flibe state. In addition, the chemistry control system will have the capability to adjust the redox potential of the salt and to correct Flibe chemistry changes induced by transmutation. The applicant also states that the test program specified in Section 4.2.4 will determine if stress corrosion cracking is a credible degradation mechanism for the environment of the KP-FHR designs. Therefore, the applicant does not propose additional material testing of 316H and ER16-8-2 for irradiation effects on stress corrosion cracking. However, a materials surveillance program and the inspection and monitoring program, as discussed in Appendix B of the TR, will be implemented and submitted in future license applications to address concerns for IASCC.

The NRC staff finds it acceptable to implement a materials surveillance program that will be submitted in future license applications for the non-power test reactor and the commercial power reactor because this program could provide sufficient information that can be used in determining any effects irradiation has on the stress corrosion cracking rate of 316H and ER16-8-2 in the KP-FHR environment. As stated in Section 5.1.3.4.2 of this SE, the NRC staff notes that the materials surveillance program should not be limited to only the first commercial power reactor, because there is limited data on the effects of irradiation on stress corrosion cracking rates in Flibe on 316H and ER16-8-2. Consistent with the discussion in Section 5.1.3.4.2 of this SE, above, this warrants implementation of a materials surveillance program for all commercial

power reactors using the KP-FHR design. In addition, the NRC staff finds it acceptable to use an inspection and monitoring program to detect cracking of the reactor vessel because the program should be capable of detecting cracking that would prevent the reactor vessel from performing its safety function. Therefore, the NRC staff is imposing a limitation and condition that the materials surveillance program and the inspection and monitoring program will be submitted in future license applications for NRC staff review and approval to verify that these programs are sufficient to address irradiation-affected stress corrosion cracking of the reactor vessel (see Limitation and Condition 14).

5.1.4 Quality Assurance

Section 1 of the TR states that the non-power test reactor application is implementing a quality assurance program based on ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors," (ANSI/ANS-15.8), which is endorsed by NRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research and Test Reactors." The NRC staff finds it acceptable to use ANSI/ANS-15.8-1995 for material testing that will only be used to support the non-power test reactor. The NRC staff notes that Revision 4 of the TR does not specify if material testing related to safety-related components will be conducted under a program that complies with the requirements of 10 CFR 50 Appendix B, as stated in previous revisions of the TR. The quality and accuracy of material testing results that could be used for the commercial power reactor must be confirmed when used to address potential materials reliability and environmental compatibility of safety-related components. Therefore, the NRC staff is imposing a limitation and condition that material testing will be conducted under a quality assurance program that complies with the requirements of 10 CFR 50 Appendix B to confirm the quality of the data obtained during the material testing that will be used for the commercial power reactor (see Limitation and Condition 15).

5.2 Evaluation Summary

The NRC staff finds that the material qualification methodology for 316H and ER16-8-2 materials in Section 4 of the TR satisfy, in part, the PDCs 14 and 31 for the KP-FHR designs and is acceptable, subject to the Limitations and Conditions found in Section 6.0 of the NRC staff's SE below. The NRC staff finds that testing at the normal operating and postulated accident temperatures, and in both Nominal Flibe and Redox Controlled Flibe, to validate the degradation of 316H and ER16-8-2 material, is acceptable since the testing duplicates the operating environment that the material will experience in the KP-FHR designs. The NRC staff also finds it acceptable that the material test samples will include not only the 316H base metal and associated ER16-8-2 weld metal, but the [REDACTED]

[REDACTED]. The NRC staff also finds that there is reasonable assurance that the degradation mechanisms to be tested as described in Section 4 of the TR include the appropriate environmental degradation mechanisms for the KP-FHR designs based on the current research and testing information provided in the TR and in References 11 and 12 of this SE. These references discuss topics such as corrosion, environmentally assisted cracking, and the effects of irradiation on materials, and their applicability in molten salt environments.

The staff has reasonable assurance the qualification program meets the requirements listed in Section 4.1 described above, as they relate to the qualification of 316H and ER16-8-2 in the

Flibe environment, because the TR describes the use of generally accepted engineering standards, unique safety features, novel design features, and the relation of facility design to the PDC.

6.0 LIMITATIONS AND CONDITIONS

An applicant may reference the TR only if the applicant demonstrates compliance with the following limitations and conditions:

1. **(Section 1.0)** As stated by Kairos in the TR, NRC staff review and approval of only Section 4 of the TR was requested. Therefore, KP-FHR designs referencing this TR may only use this TR for purposes related to the information on 316H and ER16-8-2 material found in Section 4 of the TR, subject to the specific limitations and conditions found in the NRC staff SE below. All other information related to the 316H and ER16-8-2 material will be evaluated in separate documents and licensing actions.
2. **(Sections 1.1.3.2 and 5.1)** The environmental effects qualification testing for the KP-FHR designs in this TR can only be used for other components with environments that are bounded by the environment the reactor vessel would experience and are used in this TR. For example, other components that would have Flibe on one side of the metallic material and another salt on the other side of the metallic material, or higher irradiation levels than those specified in the TR, etc. would not be bounded by this TR.
3. If the time and temperature for both normal operations and postulated accident conditions change for the KP-FHR designs, they must still be bounded by the NRC staff-endorsed ranges found in Table 2 of Regulatory Guide 1.87 for 316H, or an adequate justification must be provided for NRC staff review and approval for why the values outside of the endorsed ranges are acceptable.
4. **(Section 4.2.1)** ER16-8-2 material must be qualified to a temperature of [REDACTED] in accordance with the requirements of ASME Code, Section III, Division 5, and for a time that bounds the postulated accident conditions and be approved by the NRC staff.
5. **(Section 1.1)** Because there is information that has not yet been developed and/or reviewed as part of this TR, KP-FHR designs referencing this TR must provide information that completely and accurately describes the design of the reactor coolant boundary (and associated systems) and any associated functions it is credited to perform for NRC staff review and approval. As stated in the TR, if key design features of the KP-FHR designs change, or if new or revised regulations are issued that impact descriptions and conclusions in this TR, these changes would be reconciled and addressed in future license application submittals. Due to the potential for design changes and new or revised regulations, KP-FHR designs referencing this TR must demonstrate that all regulatory and safety requirements related to the characteristics of the metallic materials are met when considering the final design of the KP-FHR.
6. **(Section 4.1)** As presented in the TR, there are key design parameters without which the proposed reactor coolant boundary design and associated properties may not be supported. Therefore, KP-FHR designs referencing this TR must have the following:

- Flibe Salt temperatures of 550°C-650°C
- An intermediate salt coolant loop for the commercial reactor
- A Primary Heat Transport System that rejects heat to the air in lieu of an intermediate coolant loop for the non-power test reactor
- Non-power test reactor lifetime of a maximum of 5 years (1 year commissioning + 4 years operation) and commercial power reactor lifetime of a maximum of [REDACTED]
- “Near-atmospheric” primary coolant pressures
- End of life irradiation of less than 0.1 dpa

These key design parameters of the KP-FHR designs, if changed, could necessitate the modification of, or addition to, the testing program.

7. (Tables 12, 13, 14, 15 and 16) If the postulated accident conditions [REDACTED], the test temperature and time for the associated material testing in Section 4 of the TR must be increased to [REDACTED].
8. (Section 4.2.3.3 and Table 13) The impurity effects testing on 316H and ER16-8-2 must include the potential loss of Flibe chemistry control from both air ingress and intermediate salt loop ingress based on the safety analysis reports. An applicant referencing this TR must demonstrate that any potential impurity ingress (including postulated accidents) in the KP-FHR designs is bound by the testing performed as part of this TR.
9. (Section 4.2.3.2, Tables 13 and 14) An applicant referencing this TR must demonstrate that the Nominal Flibe salt composition used in the KP-FHR designs is consistent with the Nominal Flibe salt composition used in these tests including initial impurities in the salt.
10. Section 4.2.3.2, Tables 13 and 14) An applicant referencing this topical report must demonstrate that the salt compositions (with reducing agent additions and impurities from postulated accident scenarios) tested in this program bound any potential salt compositions for the KP-FHR reactor designs.
11. (Section 4.2.5) In order to address phase formation embrittlement for the KP-FHR designs an applicant must show that testing bounds potential design conditions [REDACTED] and that if a secondary phase is detected during testing, the effects on mechanical properties of 316H and ER16-8-2 must be quantified via testing and approved by the NRC staff.
12. (Section 4.2.5 and Table 11) The applicant will assess thermal cycling/stripping in future licensing submittals by minimizing the thermal gradients via appropriate design and operating conditions of the KP-FHR designs based on analysis.
13. (Section 4.2.6.1) Testing for irradiation-induced embrittlement of ER16-8-2, 316H, and the associated heat affected zone of 316H must be performed that bounds the environment representative of the KP-FHR designs, including the expected irradiation

damage (dpa) and helium content. The program describing this testing must be submitted in future license applications for NRC staff review and approval to verify this testing program is sufficient to address irradiation-induced embrittlement of the reactor vessel.

14. **(Sections 4.2.6.2 and 4.2.6.3)** As described in Sections 4.6.2.2 and 4.2.6.3 of the TR, a materials surveillance program and an inspection and monitoring program must be implemented for all non-power test reactors and commercial power reactors using KP-FHR designs to assess and monitor both irradiation-affected corrosion rates and irradiation-affected stress corrosion cracking rates of 316H and ER16-8-2 in the environment of KP-FHR designs. The materials surveillance program and the inspection and monitoring program must be submitted in future license applications for NRC staff review and approval to verify these programs are sufficient to address both irradiation-affected corrosion and irradiation-affected stress corrosion cracking of the reactor vessel.
15. **(Section 1.0)** Material testing for the commercial power reactor must be conducted under quality assurance program that meets the requirements of 10 CFR Part 50 Appendix B to confirm the quality of the data obtained during the material testing that will be used for the commercial power reactor.

7.0 CONCLUSION

Based on the evaluation above, the NRC staff concludes that Kairos has provided reasonable assurance that the information in Section 4 of the TR will satisfy, in part, KP-FHR PDCs 14 and 31 as described above, for the KP-FHR designs subject to the Limitations and Conditions in Section 6.0 of this SE. The NRC staff also concludes that the qualification program proposed by Kairos will satisfy, in part, the requirements of 10 CFR 50 and 52, as described in Section 4.1 above, with respect to contents of applications, subject to the limitations and conditions discussed above. The information provided in Section 4 of the TR establishes the material qualification methodology for environmental effects of Flibe on the 316H and ER16-8-2 structural materials to be used as a basis in future licensing actions to address potential materials reliability and environmental compatibility issues of the reactor vessel using the KP-FHR designs. The results of the planned tests, along with a description of the design, operation, inspection, and surveillance programs to manage the materials performance must be provided as part of future license application submittals.

8.0 REFERENCES

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2. Email, Nuclear Regulatory Commission Richard Rivera to John Price, "Preliminary Questions on Kairos Metallic Materials Qualification Topical Report," November 25, 2020 (ML20332A076).
3. Kairos Power LLC letter No. KP-NRC-2106-007, "KP-FHR High Temperature Metallic Materials Topical Report," KP-TR-013, Revision 1, June 30, 2021 (ML21181A386)

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4. Email, Nuclear Regulatory Commission Richard Rivera to Darrell Gardner and John Price, “Preliminary Questions on Revision 1 of Kairos Metallic Materials Qualification Topical Report,” October 13, 2021 (ML20332A076)
5. Kairos letter No. KP-NRC-2204-003, “KP-FHR High Temperature Metallic Materials Topical Report,” KPTR-013, Revision 2, dated April 26, 2022, (ML22116A247) submitting “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor,” KPTR-013, Revision 2, April 2022 (ML22116A249)
6. Kairos letter No. KP-NRC-2208-001, “KP-FHR High Temperature Metallic Materials Topical Report,” KPTR-013, Revision 3, dated August 19, 2022, (ML22231B222) submitting “Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor,” KPTR-013, Revision 3, April 2022 (ML22231B224)
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8. Kairos Power LLC, letter KP-NRC-1907-006, P. Hastings, Vice President, Regulatory Affairs and Quality, to USNRC document control desk, re: “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (Revision 1),” July 31, 2019 (ADAMS Accession No. ML19212A756).
9. US NRC, NUREG-2245, “Technical Review of the 2017 Edition of ASME Code, Section III, Division 5, “High Temperature Reactors”, dated January 2023 (ML23030B636).
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11. Stephen S. Raiman, et. al., Oak Ridge National Laboratory, TLR-RES/DE/CIB-CMB-2021-03, “Technical Assessment of Materials Compatibility in Molten Salt Reactors,” March 2021 (ADAMS Accession No. ML21084A039).
12. J. R. Keiser, P. M. Singh. M.J Lance et. al., “Interaction of Beryllium with 316H Stainless Steel in Molten Li_2BeF_4 (Flibe),” Published in Journal of Nuclear Materials, Volume 565, July 2022.

Section B



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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor

Topical Report

Revision No. 4
Document Date: September 2022

Non-Proprietary

Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
	KP-TR-013-NP-A	4	September 2022

Rev	Description of Change	Date
0	Initial Issuance	June 2020
1	Addition of Non-Power Test Reactor information and response to NRC questions dated November 25, 2020	June 2021
2	Response to NRC questions dated October 13, 2021 and elimination of the PHRS for the Non-Power Test Reactor	April 2022
3	Revise wording on contaminant testing with intermediate coolant; Revise SSRT testing temperatures and add 750°C creep test and change test to Nominal Flibe; Incorporate editorial corrections	August 2022
4	Revise <i>R</i> -ratio for fatigue testing from 0.9 to 0.1. Revise fatigue crack growth rate and added Figure 33.	September 2022

Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
	KP-TR-013-NP-A	4	September 2022

EXECUTIVE SUMMARY

This Topical Report describes the qualification plans for structural alloys used in the safety-related systems of reactors utilizing Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR) technology. The plans described herein are applicable to KP-FHR power and non-power (test) reactors. These reactors operate near atmospheric pressure, utilize high temperature fuel and use molten salt coolants to provide a high degree of passive safety.

This document describes the testing and modelling required to qualify the structural alloys used in the safety-related portion of the plants, i.e., the fluoride salt-cooled reactor system. In the reactor system, the reactor vessel is the primary safety-related component, as it serves to maintain Flibe coolant around the fuel in the reactor core. This report does not describe nor does it apply to material qualification for non-safety-related systems or components. Specifically, this report describes work to extend the ASME qualification of structural alloys to higher temperatures, to generate data in high temperature air that facilitates design, and to demonstrate environmental compatibility of the structural materials. The environmental compatibility testing for the commercial power reactor and a limited scope test plan for the shorter-lived non-power test reactor are detailed in the body of the report. Additionally, this report presents, for information, ongoing work to develop coatings and cladding and reliability and integrity management plans.

Kairos Power is requesting Nuclear Regulatory Commission review and approval of the qualification plan described in this report for metallic structural materials used in Flibe-wetted areas for safety-significant high temperature components of the reactors for use by licensing applicants under 10 CFR 50 or 10 CFR 52. This includes approval of the planned testing and analyses to address the materials reliability and environmental compatibility issues via design, operation, and inspection. The results of these planned tests and analyses, along with a description of the design and inspection program will be provided in a future license application.

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LIST OF ABBREVIATIONS

Acronym	Definition
ASME	American Society for Mechanical Engineers
ASTM	American Society for Testing and Materials
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
DPA	Displacements per Atom
DOE	Department of Energy
EAC	Environmentally Assisted Cracking
FHR	Fluoride Salt-Cooled High Temperature Reactor
FSAR	Final Safety Analysis Report
HFIR	High Flux Isotope Reactor
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
KP-FHR	Kairos Power Fluoride Salt-Cooled High Temperature Reactor
LWA	Limited Work Authorization
LWR	Light Water Reactors
MANDE	Monitoring and Non-Destructive Examination
MHTGR	Modular High Temperature Gas Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
NRC	Nuclear Regulatory Commission
OFHC	Oxygen-Free, High-Conductivity
ORNL	Oak Ridge National Laboratory
PDC	Principal Design Criteria
PIRT	Phenomena Identification and Ranking Table
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
RCL	Rotating Cage Loop
RG	Regulatory Guide
RIM	Reliability and Integrity Management
SCC	Stress Corrosion Cracking
SFR	Sodium-Cooled Fast Reactor
SSC	Structure, System, or Component
SSRT	Slow Strain Rate Testing
TRISO	Tri-Structural Isotropic

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1 INTRODUCTION (INFORMATION)

Kairos Power LLC (Kairos Power) is pursuing the design, licensing, and deployment of reactors based on Fluoride Salt-Cooled High Temperature Reactor (KP-FHR) technology. Material qualification programs designed for both a non-power test reactor (Hermes) and a commercial power reactor (KP-X) are being conducted to support these objectives. To construct these reactors, Kairos Power will rely on the use of structural alloys that are qualified for use at high temperatures in selected applications. The materials qualification program relies on materials testing, materials modeling, and inspection and monitoring programs to ensure the performance of the safety-related reactor systems. This report details the approach for safety-related metallic structural materials qualification in Flibe-wetted areas for the KP-FHR consistent with American Society of Mechanical Engineers (ASME) Section III Division 5 (Rules for Construction of Nuclear Power Plant Components, High Temperature Reactors) requirements. The non-power test reactor application is implementing a quality assurance program based on ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors," (ANSI/ANS-15.8), which is endorsed by NRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research and Test Reactors." For this reason, Kairos Power is departing from the Division 5 code requirements that would require an NQA-1 based quality assurance program for the non-power test reactor. Based on the Phenomena Identification and Ranking Table (PIRT) conducted for the commercial power reactor, this report defines a subset of work required for the non-power test reactor as well as the full set of tests for the commercial power reactor. At the completion of each of these testing programs, appropriate environmental degradation factors for the non-power test reactor and commercial power reactor can be set and satisfy ASME Section III Division 5 requirements.

The structural alloys for use in the safety-related Flibe-wetted areas of the reactor were selected considering their properties, commercial availability and if the material is qualified for use via ASME Section III Division 5. These rules for construction require demonstration of the environmental compatibility of the structural materials. A PIRT-type process as described in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" (Reference 1) was used to identify significant degradation phenomena and to develop the testing and modelling qualification presented in this report. Based on the PIRT conducted, a subset of work required for the non-power test reactor as well as the full set of tests required for the commercial power reactor are provided herein.

The design of the safety-related Flibe-wetted areas of the reactor does not require the application of cladding or coatings [[

]]. If coatings or cladding are used in the safety-related Flibe-wetted portions of the reactor, their use in the design will be in a manner consistent with ASME Code rules. For example, ASME Section III, Division 5, Subsection HB, Subpart B for structural load carrying Class A materials (Reference 2).

This report also presents an overview of an Inspection and Monitoring program (non-power test reactor and commercial power reactor) and a Reliability & Integrity Management (RIM) program (commercial power reactor) for information. The Inspection and Monitoring program further ensures material and component performance will be attained. These efforts will involve on-line monitoring systems as well as periodic inspections. The Inspection and Monitoring program for the non-power test reactor will be described as part of the operating license application. The RIM program is an integral part of nuclear component life cycle management. The unique physical features of high temperature reactors such as the

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KP-FHR presents a new paradigm for RIM that has required the Code to develop new approaches. The new approach being implemented as ASME Section XI, Division 2, “Reliability”, applies to any type of reactor design and was published for the first time in the 2019 Edition. A RIM program will be described as part of the operating licensing application for a commercial power reactor.

1.1 DESIGN OF THE KP-FHR

To facilitate NRC review and approval of this report, design features considered essential to the KP-FHR technology are provided in this section. These key features are not expected to change during the ongoing detailed design work by Kairos Power and provide the basis to support the safety review. Should fundamental changes occur to these design features or revised regulations be promulgated that affect the conclusions in this report, such changes will be reconciled and addressed in future license application submittals.

1.1.1 Design Overview

The KP-FHR is a U.S.-developed Generation IV advanced reactor technology. In the last decade, U.S. National Laboratories and Universities have developed conceptual Fluoride Salt-Cooled High Temperature Reactor (FHR) designs with different fuel geometries, core configurations, heat transport systems, power cycles, and power levels. More recently, the University of California at Berkeley developed the Mark 1 pebble-bed FHR, incorporating lessons learned from the previous decade of designs (Reference 3). Kairos Power has built on the foundation laid by Department of Energy (DOE)-sponsored, University-led Integrated Research Projects to develop the KP-FHR for both a non-power test reactor (Hermes) and a commercial power reactor (KP-X). Although not intended to support the findings necessary to approve this report, additional design description information is provided in the “Design Overview of the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” Technical Report (Reference 4).

1.1.2 Design Background

Kairos Power is developing both a non-power test reactor and a commercial power generation reactor based on KP-FHR technology. The operating parameters discussed in this topical report will apply to both reactor classes. The non-power test reactor will operate at lower power level and potentially lower temperatures than the power reactor. For the purposes of the metallic material qualification, the operating parameters for the power reactor are considered to bound those that will exist in the test reactor. One difference that will have a bearing on data needed to qualify the metallic material is the expected lifetime of the safety-related flibe-wetted components in the reactor. The non-power reactor component lifetime is expected to be limited to 5 years, while the component lifetime in the power reactor will be [[]]. The key features of the power and non-power reactors are compared in Table 1.

1.1.3 Key Features

The KP-FHR technology integrates key design features and material choices into a physically compact, intrinsically safe, high temperature reactor which will be built with existing, industrially proven materials. Key design features of the KP-FHR include the use of high temperature fuel, high boiling point molten salt coolants, and low-pressure operation. This combination of the Tri-Structural Isotropic (TRISO) particle fuel, stable high boiling temperature fluoride salt coolant, and low operating stresses results in a robust

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reactor design with intrinsic passive safety. Notably, the reactor vessel is expected to see relatively low levels of irradiation damage, <0.1 dpa for the lifetime of both the non-power and power reactors.

The fuel in the KP-FHR is based on the TRISO high temperature fuel. TRISO fuel is a carbon matrix coated particle fuel, originally developed for high temperature gas-cooled reactors, in a pebble-shaped fuel element. Coatings on the particle fuel provide retention of fission products to temperatures approaching 1600°C. The primary coolant that is used in safety-related systems is a mixture of lithium fluoride (LiF) and beryllium fluoride (BeF₂) salts in a ratio of approximately 2:1. This F-Li-Be based salt, i.e., ‘Flibe’ has been proven as an effective nuclear coolant in the Molten Salt Reactor Experiment (MSRE) research program and the operation of the MSRE nuclear reactor (Reference 74). Furthermore, there has been significant research into the stability and compatibility of Flibe in nuclear applications since the operation of the MSRE. The KP-FHR is a low-pressure reactor which operates with a modest overpressure (~0.2 MPa or 2 atm) in the reactor vessel head space to minimize contamination of the primary coolant. The low-pressure operation and associated low operating stresses are another key design feature of the KP-FHR. Low operating stresses help enable the use of conventional structural alloys and provides significant margin against high temperature failure modes such as creep-rupture.

1.1.3.1 Heat Transport Systems

The commercial power reactor is expected to include at least two heat transfer loops. A primary loop contains Flibe and maintains cooling in the core. Another heat transfer loop(s) removes heat from the primary system during normal operations. Figure 2 shows two heat transport loops for the power reactor and the operating temperature range (550-650°C). The non-power test reactor will include the primary loop and provide a heat rejection subsystem which will include a heat rejection radiator, heat rejection blower and associated ducting. For both the non-power and power reactor, the hot leg of the primary heat transport loop is anticipated to operate up to 650°C and the cold leg returns the Flibe to the reactor vessel at 550°C.

The KP-FHR design includes two decay heat removal systems. A system for providing decay heat removal is used following normal shutdowns and a separate passive decay heat removal system, [[]] removes decay heat in response to postulated events. Note that the passive decay heat removal system does not rely on electrical power to accomplish its safety function.

1.1.3.2 Containment Approach

The KP-FHR design uses a functional containment approach, like the Modular High Temperature Gas-Cooled Reactor (MHTGR) rather than a low-leakage, pressure-retaining containment structure that is typically used for light water reactors (LWRs). The KP-FHR functional containment safety design objective is to meet 10 CFR 50.34 (10 CFR 52.79) offsite dose requirements at the plant's exclusion area boundary with margin. A functional containment is defined in RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors” as a “barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions. RG 1.232 includes an example design criterion for the functional containment (MHTGR Criterion 16). As also stated in RG 1.232, the NRC has reviewed the functional containment concept and found it “generally acceptable,” provided

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that “appropriate performance requirements and criteria” are developed. The NRC staff has developed a proposed methodology for establishing functional containment performance criteria for non-LWRs, which is presented in SECY-18-0096, “Functional Containment Performance Criteria for Non-Light-Water-Reactors”. This SECY document has been approved by the Commission.

The functional containment approach for the KP-FHR is to control radionuclides primarily at their source within the coated fuel particle under normal operations and accident conditions without requiring active design features or operator actions. The KP-FHR design relies primarily on the multiple barriers within the TRISO fuel particles and fuel pebble to ensure that the dose at the site boundary (from postulated accidents) meets regulatory limits. Additionally, in the KP-FHR (but not in MHTGR designs), the molten salt coolant serves as an additional barrier providing retention of fission products that could escape the fuel particle and fuel pebble barriers. This additional retention barrier is a key feature of the enhanced safety and reduced source term in the KP-FHR. To enable fission product retention of the Flibe coolant, the reactor vessel must retain the coolant around the fuel pebbles. Thus, the reactor vessel is considered to be a safety-related structure. [[

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1.1.3.3 Reactor Vessel

The anticipated design of the KP-FHR reactor vessel is based on a vertical cylinder with bottom and top heads. The vessel is expected to be constructed from materials that are qualified by the ASME Section III. The reactor vessel serves as part of the reactor coolant boundary and supports and interfaces with other systems such as rod control, pebble handling, and heat removal systems. The reactor vessel will be designed to withstand the operational loads imparted on it by the core structures, fuel, and coolant. Additionally, the reactor vessel will be of sufficient strength and resiliency to withstand off-nominal conditions required by ASME Section III Division 5 Level B, C, and D Service Conditions (Reference 37).

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1.2 REGULATORY INFORMATION

1.2.1 Regulations Relevant to the KP-FHR Material Qualification

The KP-FHR is anticipated to be licensed under Title 10 of the Code of Federal Regulations (10 CFR) using a licensing pathway provided in Part 50 or Part 52. Applicants for construction permits for facilities licensed under 10 CFR 50 are required to provide a Preliminary Safety Analysis Report (PSAR), which provides a safety assessment of the facility in accordance with 10 CFR 50.34(a). Applicants for a Limited Work authorization (LWA) are required to submit a safety analysis that meets 10 CFR 50.34 for the scope of the LWA per 10 CFR 50.10(d)(3)(i). Subsections within 10 CFR 50.34(a) relevant to the requirement to describe design characteristics of the KP-FHR high temperature materials are listed below (note these are required to be updated as part of the operating license application in the Final Safety Evaluation Report (FSAR) per 10 CFR 50.34(b)(4)):

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50.34(a)(1)(ii)(B) The extent to which generally accepted engineering standards are applied to the design of the reactor.

50.34(a)(1)(ii)(C) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials.

50.34(a)(2) A summary description and discussion of the facility, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

50.34(a)(3)(ii) The preliminary design of the facility including the design bases and the relation of the design bases to the principal design criteria.

Similarly, applicants for combined licenses for facilities licensed under 10 CFR 52 are required to provide a FSAR which provides a safety assessment of the facility in accordance with 10 CFR 52.79. Subsections relevant to the design and performance of high temperature materials are as follows:

52.79(a)(2) A description and analysis of the structures, systems, and components of the facility with emphasis upon performance requirements, the bases, with technical justification therefor, upon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. It is expected that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products. The descriptions shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations. Items such as the reactor core, reactor coolant system, instrumentation and control systems, electrical systems, containment system, other engineered safety features, auxiliary and emergency systems, power conversion systems, radioactive waste handling systems, and fuel handling systems shall be discussed insofar as they are pertinent.

52.79(a)(ii) The extent to which generally accepted engineering standards are applied to the design of the reactor.

52.79(a)(2)(iii) The extent to which the reactor incorporates unique, unusual or enhanced safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials.

52.79(a)(2)(iv) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents.

52.79(a)(4)(ii) The design of the facility including the design bases and the relation of the design bases to the principal design criteria.

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Similar requirements to these are also included in 10 CFR 52.47 for Standard Design Certifications; 10 CFR 52.137 for Standard Design Approvals; and 10 CFR 52.157 for Manufacturing Licenses.

The use of Flibe salt environments is considered to represent a new and unique feature not typical of existing licensed light water reactor designs. The design and thermophysical properties of the KP-FHR reactor coolant enhances the safety of operations and reduces the probability of events [[]]. The design and thermophysical properties of the KP-FHR reactor coolant also provides additional functional containment protection, beyond that provided by the TRISO fuel particle, by absorbing fission products that escape the TRISO protective layer. This design feature reduces the probability of accidental release of radioactive materials. The specification limits and thermophysical properties of the reactor coolant for the KP-FHR are provided in the Kairos Power Topical Report, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” (Reference 6). This report describes the qualification and testing methods for the metallic structural materials in the high temperature Flibe salt environments for use in the Flibe-wetted areas containing safety-related high temperature components of the KP-FHR. As such, qualification of these materials using the methodology described in this report supports conformance, in part, to 10 CFR Part 50, Sections 50.34(a)(1)(ii)(C), 50.34(a)(2), 10 CFR 50.34(b)(4); and to 10 CFR Part 52, Sections 52.79(a)(2) and equivalent regulations in 52.47, 10 CFR 52.137, and 10 CFR 52.157.

1.2.2 Principal Design Criteria that are Relevant to the KP-FHR Material Qualification

Facilities licensed under 10 CFR Part 50 are also required to describe Principal Design Criteria (PDC) in their safety analysis reports supporting a construction permit and operating license application as described in 10 CFR 50.34(a)(3)(i). Likewise, applicants for standard design certifications, combined licenses, standard design approvals, and manufacturing licenses must include the PDC for a facility as described in 10 CFR 52.47(a)(3)(i), 10 CFR 52.79(a)(4)(i), 10 CFR 52.137(a)(3)(i), and 10 CFR 52.157(a).

The PDC for the KP-FHR have been established in the Kairos Power Topical Report, “Principal Design Criteria for the Kairos Power Fluoride Salt Cooled High Temperature Reactor” (Reference 7). The specific PDC in this report, which rely on or credit the design and performance of high temperature metallic structural materials include PDCs 14 and 31. These PDCs are discussed below.

The design and performance of high temperature structural alloys relates to conformance with PDC 14 because the materials used in the KP-FHR must ensure that they do not fail. The PDC states:

The safety-significant elements of the reactor coolant boundary are designed, fabricated, erected, and tested such that they have an extremely low probability of abnormal leakage, of rapidly propagating failure, and gross rupture.

The design and performance of high temperature metallic structural materials is relative to demonstrating conformance to PDC 31 because the materials used in the KP-FHR must ensure that they are not unduly stressed under operating, maintenance, testing, and postulated accidents. PDC 31 states:

The safety significant elements of the reactor coolant boundary are designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and

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postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures, service degradation of material properties, creep, fatigue, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions, and the uncertainties in determining: (1) material properties, (2) the effects of irradiation and coolant composition, including contaminants and reaction products, on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

Corrosion and Environmentally Assisted Cracking are important considerations for maintaining the integrity of the safety-significant portions of the reactor coolant boundary. Demonstration, through qualification, of the acceptability of the metallic structural materials used in the safety-significant portions of the reactor coolant boundary is a key element in establishing conformance to PDC 14 and PDC 31. The qualification requirements described in Sections 3 and 4 of this report, partially satisfy PDC 14 and PDC 31. A description of how the remaining portions of these PDC are satisfied will be provided in safety analysis reports submitted with licensing applications for the KP-FHR.

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2 STRUCTURAL ALLOYS (INFORMATION)

2.1 BACKGROUND

Ductile, face-centered-cubic iron and nickel-based alloys (i.e., ‘austenitic’ alloys) are commonly used structural materials in light water reactors due to their combination of strength, toughness, and corrosion-resistance. Light Water Reactor (LWR) operation involves modest temperatures (215-345°C) but relatively high operating pressures (~7 MPa for BWR’s and 16 MPa for PWR’s). These temperatures translate into homologous temperatures (T_H)¹ of ~0.27-0.36 for the structural materials. These homologous temperatures are low enough such that solid state diffusion rates are slow and many degradation phenomena (e.g., alloy phase stability, creep, etc.) are of limited consequence.

Like LWR’s, the KP-FHR (non-power test reactor and commercial power reactor) designs will use iron and nickel-based alloys for metallic structural components but at higher temperatures, lower pressure, and in different environments. Specifically, the design of safety-significant components of the KP-FHR reactors will use austenitic alloys at homologous temperatures [[]] in molten Flibe salt. These conditions require more consideration of high temperature material phenomena (e.g., thermal creep deformation) and, like water reactors, the molten salt coolants will require assessment for potentially susceptibility to Environmentally Assisted Cracking. For comparison, the approximate operating pressures and temperatures of LWR’s, high temperature gas reactors HTGR’s and sodium-cooled fast reactors (SFR’s) and the KP-FHR is given in Figure 3. As shown, the KP-FHR will operate at significantly lower pressures than the BWR’s, PWR’s and high temperature gas reactors and at comparable pressures but somewhat higher temperatures than SFR’s.

2.2 STRUCTURAL ALLOY SELECTION

The design of the KP-FHR reactor coolant boundary will be constructed from alloys qualified (or near qualification) by the ASME Code. Currently in ASME, Section III, Division 5, there are only a few alloys that are suitable for temperatures $\geq 600^\circ\text{C}$. These include the austenitic Alloys 304H, 316H, 800H, and 617. These four alloys have been assessed along with a modified version of Hastelloy N, the DOE developed Alloy 709, and the stainless-steel weld filler metal ER16-8-2 to both down-select candidate structural allows and to begin to identify any gaps in material availability or performance. These 7 alloys were ranked based on ten criteria:

- Status of ASME Code Qualification
- Mechanical and Physical Properties
- Experience with Molten Salts
- Experience in Nuclear Reactor Systems
- Technical Maturity
- Ability to Procure the Alloy in a Wide Variety of Product Forms
- Ease of Fabrication and Existence of a Matching Weld Filler Metal
- Environmental Compatibility of the Alloy with the KP-FHR Environments
- Degree of Regulatory Acceptance of the Alloy for use in Nuclear Systems

¹ Homologous temperature is defined as the temperature of interest divided by the melting point of the pure element that that alloy is based on in absolute units.

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- Cost of the Alloy

A comparison of these rankings is provided in Table 2. As shown, the ranking for each category were assigned on a scale of 1 to 5 with a high rank (1 or a blue filled circle) being the most desirable and a low rank (5 or an open circle) being the least desirable. A summary of the factors that influenced eliminating the other structural alloys are provided below.

Alloy 304H is similar in composition and in many attributes to Alloy 316H. However, Alloy 304H displays notably lower allowable creep stresses at high temperatures. The benefits of Alloy 304H relative to Alloy 316H are few (e.g., marginally lower cost) and do not provide compelling reasons to select this alloy over Alloy 316H. Lastly, available data indicate higher corrosion rates for Alloy 304 as compared to Alloy 316 in Flibe (Reference 34). For these reasons, 304H was eliminated from consideration in favor of the more capable Alloy 316H.

Alloy 800H is often used in high temperature applications that require corrosion resistance. However, Alloy 800H is less creep-resistant than Alloy 316H and contains higher levels of chromium (~21 wt.% Cr vs. ~17 wt.%). Higher chromium levels are undesirable for corrosion-resistance in Flibe. Furthermore, Alloy 800H does not have a matching weld filler metal but is often welded with high chromium nickel-based alloys such as EN82H. The higher nickel in Alloy 800H/EN82 relative to Alloy 316H/ER16-8-2 is less desirable due to the potential transmutation of nickel to helium, which will adversely affect irradiation embrittlement. For these reasons, Alloy 800H is ranked lower than Alloy 316H stainless steel.

Alloy 617 was recently added to ASME Section III, Division 5 and possesses superior high temperature strength and creep resistance relative to Alloy 316H. However, the alloy contains a large amount of cobalt (10-15 wt.%) which can undergo undesirable neutron activation. The high strength of Alloy 617, while desirable, is not required for the low pressure KP-FHR design. Moreover, the attractive high temperature strength can present challenges when trying to hot-form the alloy and leads to fabrication challenges. Lastly, due to the expense and limited market for Alloy 617, it is only available in limited product forms.

Hastelloy N showed excellent corrosion-resistance in the MSRE experience but was susceptible to both tellurium embrittlement and degradation by irradiation (Reference 8 and 9). For this reason, a modified grade of Hastelloy N was considered in the rankings. However, Hastelloy N is not currently approved for use by ASME in high temperature reactors and modified grades are likely different enough composition (e.g., containing several weight % of niobium) to require a full ASME qualification effort. Furthermore, it is unclear what a suitable weld filler metal for a modified Hastelloy N would be. The lack of code qualification, lack of off-the-shelf commercial availability, and high costs associated with bringing a new alloy and a weld filler metal to market are major limitations that precluded selecting a modified grade of Hastelloy N.

Alloy 709 is an advanced stainless steel being developed by the DOE for nuclear power applications. While not ASME code qualified, this effort is in progress and to date, Alloy 709 displays a desirable combination of properties with higher creep strength than Alloy 316H as well as the potential for increased resistance to irradiation damage via alloy design. Notably, welding of Alloy 709 with a weld filler metal of the same composition indicates promising properties with weld degradation factors near 1. While the lack of current code qualification and industrial supply lowers the current ranking of this alloy, it may be considered for use in future licensing applications for the KP-FHR.

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Alloy 316H and its weld filler metal ER16-8-2 possess a desirable combination of properties relative to the other candidate alloys. Alloy 316H is currently ASME code qualified, exhibits desirable mechanical properties, has demonstrated compatibility with Flibe salt, and has an extensive experience base in nuclear reactor applications. Furthermore, the alloy is technically mature with good availability, fabricability, and relatively low cost. The weld filler metal ER16-8-2 shows notable creep resistance and a high degree of weldability with Alloy 316H. Areas that require additional work for this alloy include extending the qualification of ER16-8-2 to higher temperatures (e.g., currently in the ASME code, the filler metal is limited to 650°C in the 2017 ASME Section III code), and additional research into the corrosion and environmental compatibility of these materials in Flibe. Based on this review, Alloy 316H/ER16-8-2 were selected as the metallic structural materials for safety-related components in the KP-FHR. These alloys were used as the basis for the expert panel PIRT review described in Section 4.1 which assessed environmental compatibility in Flibe salt. The remainder of the report is limited to the use and qualification of Alloy 316H/ER16-8-2 for safety-significant components in Flibe-wetted areas of the KP-FHR.

2.3 INDUSTRIAL EXPERIENCE WITH ALLOY 316H AND ITS WELD FILLER METALS

The following sections briefly describe the use of Alloy 316 in conventional nuclear reactors, advanced nuclear reactors, in non-nuclear but comparable high temperature industrial applications, and its compatibility with molten salt.

2.3.1 Conventional Nuclear Reactors

Austenitic stainless steels including Alloy 316 and Alloy 304, along with their weld filler metals, are commonly used for light water reactor internal components and corrosion-resistant cladding. Components made from these steels include fuel support structures, core barrels, flow baffle plates, and reactor vessel cladding. The low carbon variant of the alloy (i.e., the ‘L’ grade) is commonly used since high temperature strength is not limiting, but grain boundary chromium depletion (i.e., sensitization) is a concern. In light water reactors, grain boundary sensitization can result in intergranular corrosion and intergranular stress corrosion cracking if coolant chemistry is not maintained (e.g., if there is significant oxygen present in the coolant). However, Flibe salt is not oxidizing but highly reducing and its corrosion-resistance does not rely on the formation of a passive oxide film but on metallic stability in the salt. For stainless steels exposed to Flibe, the primary corrosion mechanism has been established as chromium loss (usually via grain boundary diffusion) to the coolant (Reference 10 and 11). Thus, sensitized microstructures are not inherently prone to corrosion but may be beneficial since lower chromium at the grain boundary results in less chromium lost via grain boundary diffusion.

In LWRs, irradiation can cause depletion of chromium and segregation of other elements at the grain boundaries of stainless steels and this combined with tensile stress can result in Irradiation-Assisted Stress Corrosion Cracking (IASCC) if the irradiation and stress levels are sufficient. For example, baffle-to-baffle bolts between the baffle plates in PWRs are susceptible to this degradation mechanism. However, the end-of-life irradiation doses for both the non-power test reactor and commercial power reactor are expected to be significantly lower than the dose threshold for IASCC susceptibility which is conservatively

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taken as a neutron fluence of >3 dpa for PWRs with hydrogen water chemistry. Note that the KP-FHR reactors will see <0.1 dpa at end of life.

2.3.2 Advanced Nuclear Reactors

Austenitic stainless steels, including Alloy 316 have seen extensive experience in Sodium-Cooled Fast Reactors (SFRs) (Reference 12 and 13). In SFR's, austenitic stainless steels have been used throughout the primary plant with good experience. Analogous to corrosion in molten salt, when impurities in sodium such as oxygen and hydrogen are controlled to low levels, corrosion rates are low and are governed by alloying element solubility levels in the coolant (Reference 14).

While the nickel-based alloy Hastelloy N was chosen as the structural alloy for the MSRE construction, Alloy 304 and Alloy 316 were assessed in the MSRE program for their resistance to corrosion and to tellurium embrittlement (Reference 15, 16, and 17). In loop-type corrosion tests (i.e., tests with a hot leg and a cold leg) using Flibe salt at 650°C, these austenitic stainless steels exhibited corrosion rates $\leq 25 \mu\text{m}/\text{year}$ for short exposure times (<3000 hours) which decreased with time to $\sim 8 \mu\text{m}/\text{year}$ after 3000-9000 hours exposure (Reference 16 and 17). Furthermore, when redox control of the salt was implemented (using Be metal additions), corrosion rates at 650°C were further reduced to levels estimated as $< 2 \mu\text{m}/\text{year}$ (Reference 16). While graphite can be a factor which increases corrosion rates, the data of Zheng et al., indicate this is a relatively modest $\sim 2\text{X}$ increase in corrosion rate (Reference 18).

These results indicate that corrosion will be manageable for Alloy 316 components in the KP-FHR. For example, consider a thin-walled component such as a heat exchanger tube [[

]]. Taking a reasonable, if not conservative corrosion rate of 2 microns per year (recall, Keiser et al., showed $< 2 \mu\text{m}/\text{year}$ (Reference 16) for Alloy 316 at 650°C in redox controlled Flibe) for a [[]] lifetime produces chromium loss to a depth of 0.04 mm (0.0016") or <4% of the wall thickness. It is important to note that this 0.04 mm represents a degraded layer (Cr loss), not a true reduction in wall thickness. In the chromium depleted layer, $\sim 82\%$ of the alloy remains (i.e., Alloy 316 is ~ 18 atomic % Cr).

In addition to manageable corrosion rates in Flibe salt, austenitic stainless steels also exhibit greater resistance to tellurium embrittlement than nickel-based alloys (Reference 19 and 20). The mechanism of tellurium embrittlement is well understood to be a result of the nickel – tellurium intermetallic formation (Reference 15, 21, 22, 23, and 24). Given the much lower nickel content of Alloy 316 compared to Hastelloy N, this intermetallic formation is less likely and a lower risk (Reference 25). Moreover, the KP-FHR design mitigates concern for tellurium embrittlement by the use of solid fuel and redox control of the salt (Reference 6). With the very low TRISO particle failure rate demonstrated in the DOE Advanced Gas Reactor program combined with the retention of tellurium in the fuel particle (Reference 26), the concentration of tellurium in the Flibe is expected to be significantly lower than the liquid fueled MSRE. Furthermore, the use of Be additions for redox control moves the electrochemical potential of the system away from the oxidizing regime of concern (Reference 6 and 15). For these reasons, concern for tellurium embrittlement in the KP-FHR are minimal.

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2.3.3 Other Industrial Applications of Alloy 316

Austenitic stainless steels, including type Alloy 316H are used in a wide variety of high temperature industrial applications due to their corrosion-resistance, generally desirable mechanical properties, and wide industrial availability of product forms (Reference 27). For example, Alloy 316H, its welds, and similar austenitic stainless steels (Alloy 347 and Alloy 321) are used extensively in oil and gas refinery applications at temperatures and time frames of relevance to the KP-FHR (Reference 28 and 29). For example, petroleum refining applications of stainless steels include crude distillation, fluid catalytic cracking, delayed coking, hydrotreating, catalytic reforming, hydrocracking, gas plant, amine plant, sulfuric acid alkylation, and sour water stripper systems.

Furthermore, Alloy 316H and its weld metals are used in other industries near the time and temperature of the KP-FHR. Figure 4 illustrates the intended operation of the KP-FHR in the blue box ([[]] at 550°C-650°C), relative to the NIMS creep database (gray box) and selected high temperature, long life oil and gas refinery (Fluid Catalytic Cracking Units and Cyclone) components; the typical operating temperatures and service life of these components is estimated from (Reference 29 and 76). As shown, the KP-FHR is designed to operate at somewhat lower temperature and longer times than components in the oil and gas industry. However, it is important to note that (1) there is overlap in the time/temperature ranges of experience and (2) many oil and gas components operate at higher stresses and are limited by different environmental degradation phenomena than those of the KP-FHR. For example, Fluid Catalytic Cracking Units are typically exposed to strongly carburizing gaseous environments that can limit component life and rapid temperature cycles which generate appreciable thermal stresses (Reference 29 and 76). Neither condition is pertinent to the KP-FHR.

2.3.4 Compatibility with Molten Salts

In reducing salts, Alloy 316 is used in the pyro-processing of spent nuclear fuels. In that technology, chloride-based salts are used to convert oxide based nuclear fuel back to their metallic form (Reference 30, 31, and 32). In pyro-processing systems, austenitic stainless steels are used as structural alloys and generally display excellent corrosion-resistance as long as the salt is relatively free from oxidizing impurities (Reference 33). In addition to these industrial applications, there are significant laboratory data to support the use of Alloy 316 as a structural alloy in molten Flibe (Reference 16, 17, 18 and 34).

Some corrosion data for Alloy 316 in Flibe salt are shown in Figure 17, which shows corrosion depth versus time from Reference 18 and weight change versus time from the work of Keiser and Devan (Reference 16). As shown, the corrosion rate decreases with exposure time, likely with the square root of time as diffusional transport of chromium in the alloy limits the corrosion rate. As shown in the data of Zheng (top plot) the corrosion rates are on the order of ~80 $\mu\text{m}/\text{year}$ at short exposure times (~1000 hours) and decrease with the square root of time (~21 $\mu\text{m}/\text{year}$ at 6000 hours). Additionally, the work of Keiser shows the significant benefit of Beryllium metal additions, which effectively scavenge oxidizing impurities and reduce corrosion rates.

Austenitic stainless steel is also compatible with Flibe and graphite as shown by the work of Zheng et al. (Reference 18). Those researchers performed 1:1 experiments with and without graphite in Flibe at 700°C and showed about a modest increase in corrosion depth with graphite Figure 18 – top plot). Using the

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Reference 18, data to compare predicted corrosion rates versus time, indicates graphite increases the corrosion rate approximately 2x (Reference 18). One way in which graphite likely increases the corrosion rate is by reaction with metallic chromium in the salt to form chromium rich carbides. Reaction of chromium ions in the salt to form carbides likely act to decrease the surface concentration of chromium and drive solid state diffusion. Note that in the corrosion testing programs proposed for the commercial power generation reactor and non-power test reactor, the test systems will incorporate large surface areas of graphite to capture these effects as detailed in Section 4.2.3.1.

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3 AIR TESTING AND FINITE ELEMENT ANALYSES (INFORMATION)

For the design and licensing of KP-FHR reactors to the ASME code, high temperature material property data and subsequent analyses are desired. These testing and analysis efforts can be grouped as: (1) testing to extend the code qualification of ER16-8-2 weld metal up to [[]] to match the current qualification of Alloy 316H base metal and (2) testing and analyses required to support novel, high temperature design methods including elastic-plastic and inelastic design per the ASME Section III Division 5. [[

]]

3.1 TESTING REQUIRED FOR ASME CODE EXTENSION

ER16-8-2 weld filler metal is currently qualified up to 650°C in the ASME code while Alloy 316H is qualified to 816°C (Reference 37). The KP-FHR reactor vessel operates at approximately 550°C during nominal operations but could experience temperatures up to [[]] for short durations during postulated events [[

]]. Thus, an extension of the ASME Section III code qualification for ER16-8-2 weldments will be required as described in the following paragraphs to develop a Code Case introducing stress rupture factors for Alloy 316 weldments with ER16-8-2 filler metal for temperatures between 650°C and [[]].

The types of mechanical testing that are necessary to develop a Code Case for extending the stress rupture factors for Alloy 316 weldments with ER16-8-2 filler metal are described in ASME Section III Division 5, Non-Mandatory Appendix Y (Reference 37). The methods of testing that are required for such weldments as specified in Appendix Y are the ASTM E21 Elevated Temperature Tensile Testing, ASTM E2714 Creep-Fatigue Testing, and ASTM E139 Creep-Rupture Testing. In order to meet those requirements, the planned testing is detailed in Table 3 where the number in parentheses (X) indicates the number of test samples to be tested at that condition (i.e., (2) = two replicate tests).

The welds to support these tests have been fabricated utilizing Alloy 316H base metal² and ER16-8-2 weld filler metals, and the gas-tungsten-arc welding process. The Certified Material Reports for these materials are provided in Appendix F and Figure 20 shows that the composition of the weld filler metal (Heat 578409) is within ASME specifications. Two types of welds have been fabricated to support testing: (1) weld pad buildups and (2) v-groove weldments that use Alloy 316H siderails. Examples of these welds are provided in Figure 19. Note that the proposed Code Case to extend use of ER16-8-2 up to [[]] will be based on testing of a single heat (Heat # 578409 Lot YT0384) whose material certification is provided in Appendix F. This use of a single heat of material for a code case is consistent with ASME guidelines as described in HBB-Y-2300 which refers to ASME Section II, Part D, Appendix 5-1500. This article requires “...(2) creep-rupture data for weldments made with one lot of consumables for each process intended to be used with the new base material.” The scope of the Code Case will limit the high temperature

² Note that all weld metal samples that are machined from weld pad buildups used Alloy 316L/316 base plates. The use of that material is appropriate since the samples machined far (>0.5”) from the weld/base metal interface to mitigate any dilution of the weld metal composition. For samples that contain the base metal (e.g., cross-weld samples, other heat affected zone samples, the Alloy 316H plate was used.

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applications to the Gas Tungsten Arc Welding (GTAW) process, which is the process used for fabricating the test specimens. Therefore, testing of a single heat of GTAW filler metal is sufficient to satisfy the ASME requirement.

Furthermore, these data will be assessed against larger sets of test data that encompass several heats of material to establish appropriate statistical confidence limits on the code case data. For example, a large database of ER16-8-2 weld metal was analyzed by ASME to develop a Larson Miller parameter equation for creep performance at temperatures up to 1200°F as described in ASME STP-PT-077 (Reference 40). A comparison of the ASME STP-PT-077 best estimate (solid) line and bounds (dashed lines) is given in Figure 21 which includes all available ER16-8-2 creep rupture data that has been found to date.

3.1.1 Elevated Temperature Tensile Testing

Elevated Temperature Tensile Testing was performed per ASTM E21 on all-weld-metal and cross-weld specimens at temperatures between 650°C and [[]] (1200°F – [[]]) at intervals of 38°C (100°F). These tests determined the 0.2% yield strength, ultimate tensile strength, % elongation, and % reduction in area at each temperature. The strength of the ER16-8-2 all-weld-metal at each temperature will be used to establish the creep-rupture and creep-fatigue test stresses. Note that data from these tests are available and are shown below in Figure 22. As shown, these data (triangles) are in good agreement with literature data on ER16-8-2 filler metal.

3.1.2 Creep-Fatigue Testing

Nonmandatory Appendix HBB-T of Section III, Division 5 provides a means to assess creep-fatigue of base metals, but it does not provide a dedicated means to assess creep-fatigue of weldments (Reference 37). Instead, the creep-fatigue analysis for base metals is applied to areas with welds and conservative restrictions are applied as follows (see HBB-Y-3400 of Reference 37);

“(a) limiting the inelastic accumulated strains to one-half the allowable strain limits for the base metal

“(b) limiting the allowable fatigue at weldments to one-half the design cycles allowed for the base metal

“(c) reducing the allowable creep rupture strength at weldments to a fraction of the base metal value through the weld strength rupture factor when determining time-to-rupture.”

Creep-Fatigue testing per ASTM E2714 of all-weld-metal and of cross-weld specimens is performed only to verify the adequacy of the HBB-T treatment of weldments (Reference 38). If the restrictions specified in HBB-Y-3400 bound the ASTM E2714 creep-fatigue test data, then the Non-mandatory Appendix HBB-T analysis procedures for base metal with specified restrictions for welds will have been determined to be adequate for creep-fatigue analysis of welds.

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3.1.3 Creep-Rupture Testing

Creep-Rupture tests will be performed in accordance with ASTM E139 (Reference 39). The time, temperature and load conditions for the creep-rupture tests are derived from design Service Level conditions. ASME Section III Division 5 HBB-Y-2200 allows creep-rupture curves to be extrapolated up to a factor of five from the maximum creep-rupture test duration. The maximum operating service time at each temperature is therefore divided by 5x to determine the approximate maximum test duration to assess the component life. For example, for a 100,000-hour service lifetime, a minimum test duration of 20,000 hours is sufficient to bound the operating life. The test duration and temperature can then be inserted into the appropriate creep correlation (e.g., the Larson-Miller model) to estimate the test load that will be required to produce specimen rupture at each specified time and temperature combination (Reference 40).

Testing will be performed on both all-weld-metal ER16-8-2 specimens as well as on cross-weld specimens. The rupture strength of the weld metal will be divided by the rupture strength of the base metal at each time and temperature combination to determine proposed stress rupture factors. An ASME Code balloting plan will be developed and the proposed rupture factors and supporting data will be presented to the relevant ASME Code Committees for review and approval. Progress on this extension is presently being tracked through ASME Codes & Standards Record #19-2745. Once the Code Case has been approved by ASME, then it will be presented to the NRC for approval. Once approved by the NRC then the stress rupture factors at the higher temperatures will be used in the same manner as those at the lower temperatures to determine the allowable stresses for specific temperature and time durations.

3.2 TESTING TO FACILITATE NON-POWER REACTOR AND COMMERCIAL POWER REACTOR DESIGNS

To facilitate design via the ASME Section III, Division 5, additional test data are required to calibrate and validate ASME design methodologies. A testing program to extend the qualification of the weld metal to higher temperature for the non-power and commercial power reactor designs was presented in the previous section and Table 3. For Alloy 316H stainless steel model calibration and validation that will be used for the non-power and commercial reactors, six types of tests will be performed in air: (1) tensile tests, (2) stress relaxation tests, (3) strain rate change tests (aka ‘stress dip’ tests), (4) uniaxial creep rupture, (5) notched bar creep rupture testing (aka ‘3D creep tests’) and (6) creep-fatigue testing.

3.2.1 Tensile Testing

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3.2.2 Stress Relaxation Testing

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3.2.3 Stress Dip Testing

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3.2.4 Uniaxial and Notched Bar Creep Testing

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3.2.5 Creep-Fatigue Testing

Creep-fatigue tests are listed in Table 9.

3.3 HIGH TEMPERATURE TESTING & ANALYSIS TO SUPPORT POTENTIAL DEGRADATION

As part of the Phenomena Identification Ranking Table (PIRT) process detailed in Section 4.1, some degradation phenomena (or factors that influence degradation phenomena) that are primarily driven by temperature were identified. These were: (1) degradation via stress relaxation cracking, (2) weld residual stresses, (3) other thermal stresses from operation, and (4) thermal stripping. Since those issues can be addressed via testing in air (and/or via modelling), they are addressed below.

3.3.1 Stress Relaxation Cracking

Cracking of austenitic stainless steels in the temperature range (approximately 500-700°C), aka ‘stress relaxation cracking’ has been a concern since the 1950’s (Reference 77, 78, and 79). In general, alloys of greatest concern have been the stabilized grades 347 and 321 (Reference 77, 78, 80, and 81) although types 304 and 316 stainless steel can be susceptible under conditions which produce high triaxial stresses (Reference 78, 79, 82, 83 and 84) as noted by research of Spindler et al. (Reference 86, 88 and 89).

While stress relaxation cracking of weld metal has been reported in chrome-moly steel and 347 weld metal (where Nb(C,N) precipitation leads to susceptibility), types 16-8-2 and 316 weld filler metals are noted for their resistance (Reference 78 and 79). To date, a literature search has not revealed any reported cases of reheat or stress relaxation cracking occurring in ER16-8-2 weld metal. Instead, the concern for Alloy 316H / ER16-8-2 appears to be limited to the heat affect zone and in components that are subjected to appreciable triaxial stress (Reference 79, 87, 89 and 90).

Several approaches are used to mitigate the risk of stress relaxation cracking. These include:

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- The use of Alloy 316H base metal, which is noted to be more resistant to stress relaxation cracking than the stabilized grades, as well as the use of ER16-8-2 weld filler metal, which has not been observed to exhibit stress relaxation cracking.
- Design of welds and application of welding processes and parameters which are resistant to stress relaxation cracking. Lower triaxial stresses are known to be beneficial for decreasing the risk or severity of stress relaxation cracking. For example, the bottom head to shell weld utilizes a machined weld preparation to move the weld joint from the notch formed by the shell / bottom head interface up into the shell which minimizes the triaxial stresses (Figure 23).
- Developing the capability to model weld residual stresses and to better assess the risk of any weld joints for stress relaxation cracking if needed, and
- Experiments for the non-power test reactor testing will assess the bounding triaxiality allowed by the ASME code for both Alloy 316H and Alloy 347 heat affected zones to (1) assess if this type of cracking can be triggered in Alloy 316H and (2) to compare the relative resistance of these two austenitic stainless steels. This testing will be followed by an additional study for the commercial power reactor as detailed in Table 10.

These tests follow the work of Spindler and Smith, who have used notched tensile bars to assess the effects of triaxiality and temperature on susceptibility to stress relaxation cracking (Reference 46). [[

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3.3.2 Weld Residual Stresses

Weld residual stresses are known to influence some environmental degradation phenomena like stress corrosion cracking. [[

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3.3.3 Thermal Stresses & Thermal Striping

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4 COMPATIBILITY WITH FLIBE AND IRRADIATION (APPROVAL UNLESS NOTED)

As noted above, Alloy 316H is already an acceptable material for use in high temperature reactor applications in ASME Section III. However, the code requires demonstration of the environmental and irradiation compatibility of the structural materials. For the KP-FHR safety-related systems, the environments of interest include high temperature air (external to the system) and molten Flibe salt (internal to the system), with exposure to neutron irradiation.

4.1 REVIEW OF POTENTIAL ENVIRONMENTAL AND IRRADIATION ISSUES

Due to the breadth and complexity of environmental issues, an expert panel consisting of experts from national laboratories, universities and consultants was convened to assess potential environmental issues for Alloy 316H / ER16-8-2 in each of the KP-FHR heat transport loops. This review was performed for a KP-FHR power generating reactor assuming a [[]] operational life and a secondary coolant loop that uses a different molten salt than the primary coolant loop. Based on this review, a full set of tests for the commercial power reactor was defined. A subset of tests was also developed for the lower power, shorter lifetime, non-power test reactor.

This review utilized a process based on the Phenomena Identification and Ranking Table (PIRT) methodology in NRC Regulatory Guide 1.203. Only the environmental degradation issues pertinent to potential safety-related components (exposed to Flibe and air) are summarized in this report. Component materials degradation considerations are summarized in Figure 6, which presents the Venn Diagram for the Material – Stress/Strain – Environment degradation phenomena of concern for the expert panel.

In total, there were 23 degradation phenomena assessed by the expert panel in 7 unique systems, structures, and components (SSC's). This resulted in 198 scenarios assessed by the expert panel to start, with ten scenarios added during the PIRT for 208 total rankings. Each scenario was ranked based on its importance (high, medium, low) and the degree of knowledge (high, medium, low). The PIRT rankings are shown schematically in Figure 7. Phenomena with high importance and low knowledge are the greatest priority (upper right box), followed by phenomena with high importance and medium knowledge (upper center box) and phenomena with medium importance but low knowledge (middle right box). These categories are given a numerical ranking, where Category #1 indicates that highest priority phenomena to investigate (high importance and low knowledge), Category #2 is the next important, etc. Note that each degradation phenomenon was ranked so that a total of seven, equally weighted rankings were used to develop average knowledge and importance levels.

In considering the results of the review, a conservative approach was adopted to determine which phenomena warranted future investigation. Rather than take an average ranking, phenomena were considered based on if any Expert gave it a ranking of 1 (High Importance / Low Knowledge), 2 (High Importance / Medium Knowledge), or 3 (Medium Importance / Low Knowledge). Results from those rankings are given in Figure 8. The excluded phenomena are of such low importance or high knowledge as to not warrant further consideration.

In Figure 8 the open symbols identify phenomena that will be addressed by further investigation while the 'X' symbols show the low ranking of the phenomena that will not be addressed. The degradation

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concerns that warrant further investigation are grouped into categories with corrosion related phenomena being identified by blue circles, environmentally assisted cracking by green squares, ‘other’ phenomena by gray triangles and irradiation effects by red diamonds.

The resulting phenomena to be further addressed are presented in Table 11, which summarizes the issues. Note that Table 11 only presents the degradation phenomena for safety-related components. The degradation phenomena are grouped into four categories: corrosion, environmentally assisted cracking, ‘other’ phenomena, and irradiation effects. For each category, the phenomenon of interest is listed along with a brief description and major variables that additional investigation will address.

Given the materials testing categories in Table 11 (Corrosion, Environmentally Assisted Cracking, Metallurgical Phenomena, and Irradiation Effects), the following sections outline the testing and modelling that will address those concerns. These efforts support appropriate design, operation, and inspection requirements for a [[]]] of the structural materials. Unless otherwise noted, all tests will be performed on the base materials (Alloy 316H that meets ASME Section III Division 5 compositional requirements) and on the weld filler metal (ER16-8-2).

4.2 ENVIRONMENTAL COMPATIBILITY

4.2.1 Use of the PIRT Data for NRC Licensing

The PIRT review, which identifies and ranks the appropriate environmental degradation phenomena that are applicable to safety-related components of the KP-FHR (i.e., the reactor vessel which serves the function of retaining the coolant around the fuel) was completed.

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Note that the environmental testing described below is targeted to satisfy PDC 31 for safety-related components (i.e., the Flibe-wetted reactor vessel and reactor vessel internals) which operates at approximately 550°C during normal operation and is expected to see higher temperature transients infrequently and for short time periods. To address a large range of potential operational transients as well as many accident scenarios, testing between 550-750°C and environmentally assisted cracking testing between 550-750°C is planned to be conducted. [[]]

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]] A summary of the results of transient safety analyses will be provided in the application for an operating license.

In addition to testing at or above the normal operating temperature of the reactor vessel, many of these tests encompass other aggressive testing conditions such as the use of ‘Nominal’ Flibe salt rather than redox controlled Flibe or high applied stresses and stress intensity factors relative to what the reactor vessel is expected to see. [[

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The PIRT identified two potential accident scenarios (for the commercial power reactor) that could affect the safety-related components, i.e., air ingress and intermediate coolant ingress into the Flibe salt. Note that the KP-FHR technology mitigates these concerns via design features. For example, air ingress is prevented via the hermetically sealed containment and the use of an inert gas overpressure in the reactor vessel. Similarly, intermediate coolant contamination of the Flibe is prevented via the design of the intermediate heat exchanger and by operating the Flibe at a higher pressure than the intermediate heat exchanger. Testing to better assess the effects of air and intermediate coolant contamination of the Flibe as detailed below in Section 4.2.3.3 is planned. Note that the non-power test reactor system uses a Flibe-to-air heat rejection subsystem.

A third potential accident scenario, water ingress into the Flibe has been discussed but judged not to be credible for the following reasons. There are two potential sources of water near the reactor vessel, a cavity cooling system and a decay heat removal system. Both these systems contain design features such that water ingress into the Flibe is not a credible accident scenario as described below.

The potential location of the cavity cooling relative to the reactor vessel is shown schematically below in Figure 25. As shown, cavity cooling is planned to be inside the concrete cavity wall and further separated from the reactor vessel via a steel liner. Given these design features, failures in the cavity cooling do not have a credible path to cause water ingress into the reactor vessel or heat transport system Flibe in any credible operational or event scenario.

Similarly, the decay heat removal system uses multiple design features to preclude any contact with the reactor vessel or Flibe salt. The decay heat removal system is planned to use two barriers to prevent leaks into the reactor cavity. The first barrier is the primary heat transport system piping and reactor vessel physical boundary. Outside of that there is a second wall. The intervening gas space humidity is monitored to check for leaks in the primary heat transport piping. This prevents Flibe-water interaction in two-ways:

- (1) leaks from the decay heat removal system would be identified sufficiently early to shut down the system and replace components as necessary before bulk water can enter the reactor cavity.
- (2) The outside barrier would act to protect the decay heat removal system piping from the thermal shock associated with Flibe spraying on it. The external barrier used for leak checking prevents direct spray on

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the decay heat removal system, which inhibits bulk-interaction between water and Flibe. Both barriers must fail before bulk interaction may occur.

4.2.2 Alloys and Heats to be Assessed

Potential heat-to-heat variability in environmental testing was identified in the PIRT. For the known degradation of austenitic stainless steel exposed to fluoride salts, the primary issue is loss of chromium from the grain boundaries to the salt. This grain boundary chromium loss has not exhibited heat-to-heat sensitivity but is fundamentally controlled by the solid-state diffusivity of chromium, a process that is not sensitive to minor changes in alloy composition. For example, recent evaluations show very similar corrosion rates between dual certified 304L/304 stainless steel in Flibe and 316L/316 stainless steel and with a fundamentally based prediction as shown below in Figure 26.

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4.2.3 Corrosion

Corrosion tests of prototypic materials to develop quantitative corrosion rate models for Flibe will be conducted. [[

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4.2.3.1 Testing Systems

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4.2.3.2 The Use of Compositional Analysis and Electrochemical Potential (ECP) Monitoring

Both compositional analysis of the salt and ECP monitoring are planned. [[

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4.2.3.3 Corrosion Testing

The planned corrosion testing is summarized in Table 12 which gives the purpose of the test, the materials to be tested, the environment and the approximate test temperatures and duration. For each test, the depth of chromium loss will be assessed over time to establish the governing corrosion kinetics (Equation 1) and to establish the steady state corrosion rate. Note that while the weight change of each corrosion coupon will be documented, the analytical electron microscopy is intended to be used to determine the extent of corrosion or other metallurgical changes (e.g., Cr loss depth, carbide precipitation, etc.). Additional details of the corrosion testing and an example of the planned statistical analysis of the data are provided in Appendix C.

The purpose of each test is further elaborated in Table 13 and discussed below. For most tests, the corrosion rate will be established by assessing the depth of chromium loss from the sample surface. The chromium loss depth will be determined by an appropriate analytical technique such as wavelength dispersive spectroscopy. In addition, the weight change of the corrosion coupons will be determined. The following bullets expand on the purpose of each test.

$$Cr \text{ loss depth} \propto \left(t, \sqrt{t}, \log t, \frac{1}{\log t} \text{ etc.} \right) \quad \text{Eq. 1}$$

- **Temperature:** The testing as a function of temperature in nominal Flibe for Alloy 316H and ER16-8-2 will determine the corrosion rate for each alloy and will be used as a baseline to judge subsequent separate effects testing. At each of the three planned temperatures, tests will be conducted for different times to determine the controlling kinetics and the steady state corrosion rate. The steady state rates will then be used to develop best-estimate and design-estimate predictions of corrosion rate as a function of temperature. These data will be fit to a model of the form of Equation 2 and provide a standard against which the separate effects tests described below can be quantitatively judged.

$$\text{Corrosion Rate} = A \cdot \sqrt{t} \cdot \text{EXP} \left(-\frac{Q}{RT} \right) \quad \text{Eq. 2}$$

- **Microstructural Effects:** The effects of the weld heat affected zone, long-term thermal aging, and cold work (20% via rolling) will be assessed and compared to the baseline (temperature dependent models). [[

]] This testing has been eliminated since a post weld heat treatment will not be utilized.

- **Salt Composition:** The salt composition testing will assess the effects of the impurities and redox control. The impurity testing will cover accident scenarios defined in the materials PIRT review: nitrate ingress for 168 hours and air ingress for 168 hours (i.e., scenarios 3 and 4). The conditions of the accident scenarios have not been defined at this time and will be provided in safety analysis reports submitted with a future license application. These tests will determine the effect of

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potential loss of salt chemistry control on the corrosion rate. Redox control will be investigated via separate effects testing in order to define a factor of improvement in corrosion rate relative to the nominal Flibe purity.

- **Occluded Geometry:** The intent of these tests is to investigate if a physical crevice influences the corrosion rate with and without redox control of the salt. Occluded geometry will exist on all corrosion samples due to small gaps between samples and the sample cage of RCL systems. The nominal aspect ratio of this crevice is 12 (width/depth) and the minimum aspect ratio is 3.17, based on fabrication tolerances. For a subset of samples, these creviced surfaces will be characterized and compared to fully exposed surfaces. Additionally, unloaded, pre-cracked reference samples used in SCC testing will be used to further evaluate occluded geometry effects. These reference samples will be fitted with an insert which creates long crevices. Note that screening work at ORNL on a nickel-based alloy indicates that crevice corrosion is not a concern in fluoride salt (Reference 87).
- **Erosion-Corrosion and Graphite Contact:** These tests will assess the potential effect of erosion-corrosion. Specifically, graphite particulate will be introduced into corrosion tests with flow to assess if hard particles (e.g., potentially from the graphite reflector) will significantly impact corrosion rates. In these tests, weight change of the coupon (via chromium loss depth) will be used as the primary indicator of the corrosion rate.
- **Cold Leg Occlusion:** In addition to the effect of temperature on the corrosion rate (hot leg samples), many of these tests described above will be used to assess the potential for cold leg occlusion. Cold leg occlusion will be assessed by monitoring the flow rate of salt circulating between hot and cold legs. This rate is estimated using heat flow analysis and furnace power inputs. Additionally, RCL systems will be inspected during planned shutdowns for sample exchanges, and during decommissioning and teardown to look for evidence of cold leg occlusion.

4.2.3.4 Discussion of Redox Control and Monitoring Test Systems

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4.2.4 Environmentally Assisted Cracking

Literature data for environmental degradation of both stressed and unstressed samples were recently reviewed in Reference 42 and updated in Table 20. In general, there has been little mechanical testing in molten salts and few data of relevance to the KP-FHR. In part, this is due to the difficulty of conducting in-situ mechanical testing in highly reducing molten salt.

Relatively little literature data exists for structural alloys undergoing environmentally assisted cracking in molten salts. Fluoride salts are of primary interest to the KP-FHR technology and Table 20 summarizes literature studies of EAC in molten fluoride salts. For the 10 studies shown, 7/10 are for Ni-Mo-Cr family of alloys (INOR-8 / Hastelloy N or variants) that were used in the Molten Salt Reactor Experiment (MSRE), while only two studies investigate austenitic stainless steels and there is one report of EAC in oxygen free high conductivity (OFHC) copper based on post-operation examination of the MSRE.

Evidence for intergranular stress corrosion cracking of the OFHC copper component was clear but the precise environmental and mechanical conditions that produced the cracking were not. This observation was from the post-irradiation examination of materials from the MSRE where a cup used to take salt samples was inadvertently lost, plastically deformed during attempts to retrieve it, and subsequently exposed to fuel salt for ~13,000 hours (Reference 9). The cup was recovered post-operation and showed extensive intergranular cracking and the copper was noted to be brittle. Compositional analyses via the electron microprobe and mass spectrometry indicated that the copper alloyed with nickel with local regions enriched in chromium and molybdenum, including a grain boundary second phase that was rich in chromium. No conclusions were reached as to the causes of the brittleness of the OFHC copper (Reference 9).

More extensive research into environmental compatibility of the structural alloy INOR-8 (Hastelloy N, Ni-16Mo-7Cr-4Fe) was also part of the Aircraft Reactor Experiment (ARE) and the subsequent MSRE programs. The mechanical properties of several heats of INOR-8 exposed to the ARE fuel salt 11.2NaF-41KF-45.3LiF-2.5UF₄ (mol %) were reported by Swindeman (Reference 97). In that work, tensile tests, creep tests, and stress relaxation tests were performed in both air and the fuel salt. It is notable that in creep testing, the fracture mode of the alloy was often intergranular in both salt and in air but creep rupture ductility was sufficiently high (often ~6% elongation when tests were conducted to failure) such that no environmental degradation in salt was noted (Reference 97). It is interesting to compare these results with the more recent research of Shrestha et al. on a similar Ni-16.0Mo-6.4Cr-3.6Fe alloy (Reference 98). In those creep rupture tests performed in both air and in FLiNaK salt at 700°C, there was clear evidence of environmental degradation with the salt exposed samples exhibiting ~50% shorter

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failure times and extensive intergranular cracking. Notably, the creep failure strains in both environments were high with failure occurring ~20% strain in FLiNaK and ~50% strain in air. The mechanism of the degradation in the FLiNaK was suggested to be preferential corrosion of molybdenum depleted grain boundaries (Reference 98).

One mechanism of EAC discovered in the MSRE program that has garnered some additional research is the embrittlement of nickel-based alloys by tellurium (and similar Group 16 ‘Chalcogen’ elements) and similar elements (Se, S, etc.) (References 9, 15, 20, 22, 23, 24, 99, 100 and 101). In laboratory testing, intergranular cracking via tellurium embrittlement was noted in several nickel-base alloys (Hastelloy N, Hastelloy S, Inconel 600) but not in Alloy 304 stainless steel (Reference 20). Under conditions where tellurium is plated on the surface of the alloy (oxidizing potentials), it preferentially diffuses down fast paths like grain boundaries and forms brittle intermetallic compounds (e.g., Ni_3Te_2 , CrTe , MoTe_2 , etc.) that cause intergranular cracking (References 23, 24, 102 and 103). Iron-based alloys such as 304 stainless steel appear to be more resistant to this type of embrittlement, in part due to the lesser tendency to form the Ni and Mo – rich intermetallic phases. Embrittlement of Ni-Mo-Cr alloys (INOR-8 / Hastelloy N) can be mitigated via alloying with ~1-2 wt.% niobium, which is believed to slow Te grain boundary diffusion (Reference 20). However, alloying with both Nb and Ti was shown to negate the beneficial effect of niobium (Reference 20). In addition to the embrittlement produced by intermetallic formation, first-principles work indicates that tellurium, like sulfur, also causes intrinsic grain boundary Ni-Ni bond weakening and likely promotes grain boundary decohesion (Reference 104, 105 and 106).

Environmental degradation of nickel-based alloys via sulfur embrittlement was noted in mechanical testing (Reference 99) during salt processing (Reference 25) of the MSRE program. These observations showed (1) intrinsic embrittlement of nickel or nickel alloy grain boundaries by sulfur as show by high temperature exposure and low temperature testing and (2) cracking due to the formation of the low melting point Ni_2S eutectic. In mechanical testing of Hastelloy N, exposure to a sulfur rich environment resulted in a degradation of both the tensile and creep performance of the alloy (Reference 99). Similarly, impure Flibe salt has been noted to cause cracking of nickel alloy processing and testing equipment (e.g., a salt transfer line), likely through the formation of a low melting eutectic (Reference 25 and 107). Iron based alloys like austenitic stainless steel are notably more resistant to the development of sulfur rich eutectic compositions with the lowest melting Fe-S eutectic forming approximately 1027°C while Ni_2S forms approximately 636°C.

The MSRE program also performed limited testing of austenitic stainless steels which were shown to be more resistant to both tellurium embrittlement and degradation by sulfur than INOR-8. However, austenitic stainless steels contain more chromium than INOR-8 (~17 wt.% vs. ~7 wt.%) which can result in faster corrosion rates under oxidizing conditions. An in-situ mechanical testing system was developed to support additional investigation of this phenomena which is shown schematically in Figure 32. Key features of the testing systems include:

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The in-situ mechanical testing systems will be used to conduct the slow strain rate, corrosion fatigue, stress corrosion cracking, and in-situ creep testing described below.

4.2.4.1 Slow Strain Rate Testing

Slow strain rate testing (SSRT) will be conducted in nominal Flibe to assess if Alloy 316H, ER16-8-2, and the heat affected zone of Alloy 316H are susceptible to environmentally assisted cracking in fluoride salts. The SSRT is a well-established and accepted methodology to determine susceptibility to stress corrosion initiation and crack growth (Reference 43). Testing will be conducted in accordance with ASTM guidelines outlined in ASTM G129-00 (Reference 44). The SSRT tests will be conducted on flat, pin-loaded specimens. Tests will be conducted at three different temperatures 550, 650, and 750°C, at four (4) strain rates between 1×10^{-6} - 5×10^{-8} (in/in)/sec as detailed in Table 14. Note that the 750°C tests will be run only at faster stroke rates, consistent with potential accident scenarios. In the tests, the degree of an environmental effect will be assessed by comparison of the load/stroke curves with comparable tests conducted in air as shown schematically in Figure 10. Additionally, the fracture mode of these test samples will be investigated to better assess any potential environmental damage. For the non-power test reactor, only heat affected zone samples will be tested. These samples contain all three materials of interest (weld metal, Alloy 316H heat affected zone, and unaffected Alloy 316H) within the gauge section of the tensile bar and are judged to be an efficient method to assess EAC susceptibility.

4.2.4.2 Fracture Mechanics Based Testing: Corrosion Fatigue and Stress Corrosion Cracking

In addition to the slow strain rate testing, fracture mechanics-based testing will be performed on pre-cracked samples based on established methods (Reference 45). These tests will assess prototypical materials (Alloy 316H and ER16-8-2 weld filler metal and the Alloy 316H heat affected zone) and be conducted in nominal Flibe and redox controlled Flibe as provided in Table 15. These tests will include both a corrosion fatigue portion of the test and a constant stress intensity factor portion of the test to assess stress corrosion cracking. The corrosion fatigue portion of the test will initially be at relatively high ΔK 's to produce fatigue crack growth and will subsequently shed load to both (1) determine the 'Stage II' Paris-law crack growth rate and (2) to prepare the sample for subsequent stress corrosion cracking testing.

These in-salt fatigue crack growth rates will be compared to similar data determined at temperature but in-air to assess any potential degradation, e.g., the difference between in-air vs. in-salt behavior. Example corrosion fatigue data and their comparison to air data are shown in Figure 11. At the completion of the corrosion fatigue portion of the testing, constant stress intensity factor (K_I) testing will be conducted.

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The intent of the constant K portion of these tests is to attempt to initiate stress corrosion under aggressive testing conditions and then transition to conditions that are more representative of the KP-FHR. For the non-power test reactor, only tests on heat affected zone samples will be tested. In these tests, the sample notch will be machined in the heat affected zone of the sample. A sharp flaw (i.e., the corrosion fatigue crack) is placed near the microstructures of interest (weld metal, HAZ and base metal) and allows the crack to grow in the region of highest susceptibility. This methodology has been applied in other EAC testing where intergranular stress corrosion cracking was shown to occur preferentially in the HAZ in samples of EN82/A600 HAZ/A600 (Reference 109). These tests will be followed up by dedicated tests of the base metal and the weld metal as provided in Table 15. One potential SCC mechanism (strain accelerated corrosion and subsequent intergranular cracking) and how its stress corrosion rates may evolve are provided in Figure 12.

4.2.4.3 Environmental Creep Testing

The potential for environmental degradation during creep loading will be assessed for environmentally assisted cracking. For the non-power test reactor, the potential for environmentally affected creep rates will be assessed by comparing SSRT data from air to SSRT data in both nominal Flibe and redox controlled Flibe. The SSRT tests incorporate creep effects during deformation as evidenced by changes in the flow curve as provided in Figure 10 (References 110 and 111).

For the commercial power reactor, creep testing in Flibe will be conducted as indicated in the baseline testing plan.

Creep-rupture testing in Flibe for the commercial power reactor will be conducted to further assess the compatibility of Alloy 316H, ER16-8-2 filler metal, and the Alloy 316H weld heat affected zone with the molten salt. This testing will target creep rupture times on the order of 500 hours and 2000 hours. The creep tests will be conducted at 550°C and 650°C in nominal Flibe and will assess the integrated effects of environment and stress on the materials performance. Additionally, a short time 750°C test will be conducted to cover potential accident scenarios. The stress is lower based on the higher test temperature (750°C). A pressure of 83 MPa corresponds to the best estimate 200-hour creep rupture life for 316H plate at 750°C from the Japanese NIMS Database. ‘Cross weld’ samples will be used such that the gauge section of the creep sample contains both base metal, heat affected zone and weld metal to best assess a range of materials and microstructures. These creep rupture times will be compared to data from air tests to determine any reduction in creep rupture lifetime due to the salt. Also, the samples will be characterized for chromium loss and compared to unstressed corrosion coupons. The targeted environmental creep test conditions are given in Table 16. Note that replicate tests will only be conducted if significant degradation is observed, e.g., a failure time outside of the 90% confidence interval for air test data and/or if a change in fracture mode is observed.

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4.2.5 Metallurgical Effects

The potential environmental degradation phenomena grouped into the ‘other’ category were stress relaxation cracking, phase formation embrittlement, and degradation driven by thermal cycling or by thermal gradients. Each of these phenomena will be addressed to assess the risks of each phenomenon for Alloy 316H. Assessing stress relaxation cracking involves testing in air and as discussed in Section 3.3.1 further analysis and design changes indicate that the risk of this phenomena is sufficiently low such that additional testing is not required.

Testing for phase formation embrittlement addresses the concern that some element could be picked up by the stainless-steel during exposure to Flibe (e.g., carbon or beryllium) and form a deleterious second phase. For example, near-surface carbide precipitation in Alloy 316 exposed to Flibe+ graphite has been noted by Zheng et al. (Reference 18). Similarly, when beryllium metal is coupled to nickel, iron, or stainless steel and exposed to elevated temperature, Be can diffuse into the other metal and can exacerbate corrosion rates (Reference 48). When excess Be is present in nickel, iron or similar alloys, Ni-Be precipitates can form and increase corrosion rates, possibly by generating internal stress (Reference 48 and 49). [[

]] These samples will include

at least one SSRT sample (for the non-power test reactor) and one in-situ creep sample (for the commercial power reactor) as detailed in Table 17.

Lastly, degradation of materials can be driven by thermal phenomena that are influenced by the environment. For example, poor mixing in the coolant could lead to local temperature gradients and result in unwanted thermal stresses (thermal striping). Similarly, the large thermal transients associated with draining and/or filling the reactor vessel could result in ‘ratcheting’ of the pressure vessel. However, several design features and the high Prandtl number of Flibe act to reduce the magnitude of thermal stresses (Reference 50). These phenomena are considered to be appropriately addressed via analysis and specific concerns can be mitigated via design and operational procedures without the need for testing.

4.2.6 Irradiation Effects

The PIRT review identified three irradiation-influenced phenomena that may warrant additional work; irradiation-induced embrittlement, irradiation affected corrosion, and irradiation assisted stress corrosion cracking (IASCC). The following sections describe the additional investigation activities to address irradiation effects. The results of these efforts are to establish the appropriate design, operation, and inspection requirements for the non-power test reactor and the commercial power reactor systems. The expected reactor vessel irradiation damage and helium generation are compared in Table 21. While current estimates for the displacement damage are estimated at <0.1 dpa for both reactors, the shorter-lived non-power test reactor will generate less helium than the longer-lived power reactor. A more detailed evolution of the displacement damage and helium generation for the commercial power reactor is provided in Figure 13.

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4.2.6.1 Irradiation-Induced Embrittlement

The existing published data on austenitic stainless steels indicate that tensile properties at temperatures from 550°C to 650°C are relatively unaffected by <0.1 dpa and ~10 appm of He when tested at moderate or high strain rates ($>10^{-3}\text{s}^{-1}$). For example, a compilation of tensile data in Reference 51 indicates virtually no change in yield strength or tensile elongation ≤ 0.1 dpa for several austenitic stainless steels, including Alloy 316 variants. Similarly, fracture toughness remains high in austenitic stainless steels below 0.1 dpa, with values in excess of 100 MPa $\sqrt{\text{m}}$ (Reference 52). While most fracture toughness studies focus on LWR conditions (Figure 14), those data indicate that fracture toughness remains high at ~0.1 dpa. Work by Bernard on Alloy 316H (Reference 53) and DeVries on Alloy 304 (Reference 54) at 550°C confirm that fracture toughness is high at conditions of the KP-FHR operation with J_{IC} values near 100 kJ/m². In Figure 14, the apparent increase in toughness at 0.3 dpa may be due to some irradiation-induced hardening before any appreciable loss in ductility, which is reasonable based on the tensile data of Nagae (Reference 51).

However, when testing at low strain rates, irradiation-induced embrittlement is clearly a concern where structural alloys are known to exhibit helium embrittlement. An example study of the effect of strain rate and temperature on ductility of an austenitic stainless steel is shown in Figure 15 (Reference 55). As shown in Figure 15, tensile ductility remains unaffected at strain rates $\leq 10^{-2}\text{s}^{-1}$ but slowly decreases as strain rate is lowered, especially in the temperature regime of ~500-700°C. To better assess this effect, literature-reported changes in creep properties after low-dose irradiation in Alloy 316 and Alloy 316 weld metals are summarized in Figure 16. While the data show some scatter, creep strength can decrease by up to ~30% after irradiation. Meanwhile, creep ductility is shown to either increase or decrease by up to 20% (in base metal) or 70% (in weld metal) after irradiation. These data will be used to determine the appropriate environmental degradation factor for creep life of both the non-power test reactor and the commercial power reactor vessels.

Based on these literature data, no additional testing for irradiation embrittlement of material properties is required for either the non-power test reactor or the commercial power reactor. Instead, existing data will be used to develop degradation factors. However, irradiation tests of Alloy 316H, its heat affected zone, and ER16-8-2 weld filler metal will be conducted to better quantify design margins at the relatively low irradiation damage levels of the non-power test reactor and the commercial power reactor vessels. Details of the testing program will be provided with the operating license application.

4.2.6.2 Irradiation-Affected Corrosion

Corrosion in KP-FHR could be affected by irradiation through irradiation-induced changes in the redox potential of Flibe, irradiation-induced changes in the corrosion resistance of stainless steel, or both. In water-based systems, both mechanisms (water radiolysis and defect production in stainless steel) are thought to lead to irradiation-accelerated corrosion (Reference 58). However, these mechanisms are not applicable to the KP-FHR environment. First, Flibe is highly resistant to radiolysis because of the rapid recombination of ions in the molten state. Second, while irradiation could affect the chemistry of Flibe through transmutation, the chemistry control system will have the capability to adjust the redox potential of the salt and correct changes induced by transmutations, expected to be very small. Third, irradiation-induced defect in stainless steel can lead to radiation-enhanced diffusion, which may affect corrosion, but

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because of the high operating temperature of 550°C and the dpa rate of 0.1 dpa / [[
]] the vacancy concentration is not significantly affected by irradiation, and radiation-enhanced diffusion is expected to be minimal.

Existing data indicates that irradiation effects are limited and can be both negative and positive. For example, Lei et al., show a modest increase in post-irradiation bulk corrosion rates (~3X faster) in FLiNaK salt after ~6.18 dpa irradiation with helium ions (Reference 59). In contrast, recent work by Short et al., indicates that simultaneous irradiation and corrosion in FLiNaK acts to minimize intergranular corrosion in molten salt (Reference 60 and 61). Apparently, increased near-surface vacancy concentrations from irradiation accelerates general corrosion (likely controlled via bulk diffusion) but increased intragranular vacancies promotes diffusion from grain interiors to the grain boundary, effectively lowering grain boundary corrosion rates.

Given that: (1) the only safety-related component that is subject to irradiation is the thick-walled reactor vessel, (2) the irradiation dose is quite low <0.1 dpa and (3) irradiation has shown a benefit to grain boundary corrosion (which is the primary concern), no immediate testing is planned. Instead, irradiation affected corrosion will be assessed via (1) a materials surveillance system program for the non-power test reactor and (at least the first) commercial power reactor systems and (2) an inspection and monitoring program that will assess the wall thickness of the reactor vessel. The initial plans for these programs are provided in Appendix B. Additional details of these plans will be provided in the operating license application.

4.2.6.3 Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Similar to irradiation affected corrosion, IASCC is not an expected degradation mode in the KP-FHR. The two main pathways for IASCC in water environments are radiation effects on the water chemistry and on the materials (Reference 65). In the KP-FHR environment:

- Radiolysis of Flibe is not a concern, as detailed in in Section 4.2.6.2, and no irradiation-induced changes in the corrosion potential is expected;
- The accumulated dpa in the reactor vessel of <0.1 dpa, which is lower than the lower bound of ~3 dpa for IASCC observed in pressurized water reactors (Reference 65).

Furthermore, without significant hardening in the alloys at 0.1 dpa (Reference 51), and a potential benefit to grain boundary corrosion rates (Reference 60 and 61), there is no known mechanism by which irradiation would increase susceptibility to IASCC. The testing program will assess if stress corrosion cracking can occur in unirradiated materials (Section 4.2.4). However, available evidence indicates that this is not a credible degradation mechanism under conditions relevant to the KP-FHR. Since this test program is expected to show that there is no direct concern for stress corrosion cracking and since there is no clear means by which irradiation could increase susceptibility (i.e., no expected effect on the coolant chemistry, only a small amount of hardening at 0.1 dpa), no direct IASCC testing is planned at this time. Instead, this potential concern will be addressed via both surveillance samples and an inspection and monitoring program as discussed in Appendix B.

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5 CONCLUSIONS AND LIMITATIONS

5.1 CONCLUSIONS

Alloy 316H base metal and ER16-8-2 weld filler metal have been selected as the metallic structural alloys for use in safety-significant, high temperature, Flibe-wetted component designs. This testing is being conducted to support the design and licensing of both the non-power test reactor (Hermes) and the commercial power generation reactor (KP-X). This testing is focused on structural alloys 316H and ER16-8-2 for the reactor vessel, which was determined to be the primary safety-related component of interest, as it serves to retain the Flibe coolant (a fission product barrier) around the fuel pebbles.

The materials testing consists of two major efforts: (1) testing in high temperature air to support ASME design (submitted for information) and (2) testing in molten Flibe salt to account for potential environmental degradation (submitted for review and approval). Testing to support design includes work to extend the code qualification of ER16-8-2 weld metal up to 816°C to match the current qualification of Alloy 316H base metal as well as testing and analyses required to support elastic-plastic and inelastic design per the ASME Code Section III, Division 5.

The environmental effects testing plan detailed in this report is based on an independent Expert Panel PIRT review for the operation of the commercial power generating reactor (KP-X). As detailed herein, the scope of testing for the non-power test reactor is reduced, based on the lower power and shorter time of operation relative to the commercial power reactor. While not required in the KP-FHR design for structural performance considerations, [[
]] . Appendix A of this report details cladding and coating materials that could be used with safety-related high temperature components of the KP-FHR. Such coatings do not affect structural performance of the underlying base metals and will be used consistent with ASME Section III code requirements.

Kairos Power is requesting Nuclear Regulatory Commission review and approval of the environmental effects testing plan described in this report for metallic structural materials used in safety-related Flibe-wetted areas high temperature components of the reactors for use by licensing applicants under 10 CFR 50 or 10 CFR 52. This includes approval of the planned testing and analyses to address potential materials reliability and environmental compatibility issues via design, operation, and inspection. The reactor vessel is credited for maintaining its integrity and retaining fluid to keep the fuel covered in salt during all normal operations and postulated events. The qualification plan for these materials support conformance, in part, to PDC 14 and PDC 31. The qualification plan intends to qualify the reactor vessel and safety-related Flibe-wetted areas and to maintain its integrity under the expected environmental conditions of the KP-FHR. The results of the planned tests and analyses, along with a description of the design and inspection program will be provided in a future license application.

5.2 LIMITATIONS

This report is limited to the qualification of metallic structural materials (Alloy 316H and ER16-8-2) for safety-significant, high temperature components in Flibe-wetted areas of the KP-FHR reactors.

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Table 1. Summary of Key Parameters for the Power Reactor and the Non-Power Test Reactor

Parameter	Power Reactor	Non-Power Test Reactor
Reactor Description	Low pressure, fluoride salt-cooled high temperature reactor (FHR)	
Core Configuration	Pebble bed core, graphite reflector, and enriched Flibe molten salt coolant	
Physical Dimensions	Reactor Vessel is ~4 m diameter, ~6 m height	Reactor Vessel is ~2.5 m diameter, ~4 m height
Reactor Thermal Power	320 MW _{th}	35 MW _{th}
Primary Heat Transport System	Flibe Salt, 550°C-650°C, ~0.2 MPa, ~0.11-0.15 m/s	
Intermediate Heat Transport System	Intermediate Coolant, <0.2 MPa, 360°C-600°C	None. Primary Heat Transport System rejects heat to the (air) heat rejection subsystem
Power Conversion System	300°C-585°C, steam ~19 MPa	None. Primary Heat Transport System rejects heat to the (air) heat rejection subsystem
Material for Safety-Related Structures	Alloy 316H and ER16-8-2 (ASME Section III, Division 5, approved)	
Lifetime	[[]]	≤5 years (1 year commissioning + 4 years operation)
End of Life Irradiation of Reactor Vessel	<0.1 dpa	

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Table 2. Ranking of Structural Alloys for FHR Applications

	304H	316H	ER16-8-2 Filler Metal	800H	617	Modified Hastelloy N	709
Code Qualification							
Mechanical & Physical Properties							
Experience with Molten Salts							
Experience in Rx Systems							
Technical Maturity							
Ability to Procure							
Fabrication Considerations							
Environmental Compatibility							
Regulatory Acceptance							
Cost							
Summary	Lower strength than Alloy 316H, no compelling advantage	Best combination of properties of current ASME approved alloys. Filler metal matches base properties	Potential application, esp. in nitrate salt. No matching filler metal	High Cobalt undesirable. Ductility decrease with aging	Lack of Code Qualification and Supply. No matching filler metal	Desirable for future improvements. 709 Filler metal matches properties	
Key		Little / no work	Reasonable Work	Significant work required	Major work required	Work not initiated, major effort	

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Table 3. Summary of Tests to Extend the ASME Qualification of ER16-8-2 to 816°C

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 4. Summary of Tensile Tests to Support Non-Power Test Reactor Design

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 5. Summary of Stress Relaxation Tests to Support Non-Power Test Reactor Design

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 6. Summary of Strain Rate Change (aka ‘stress dip’) Tests to Support Non-Power Test Reactor Design

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Table 7. Summary of Uniaxial Creep Tests to Support Non-Power Test Reactor Design

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
	KP-TR-013-NP-A	4	September 2022

Table 8. Summary of Notched Bar Creep Tests to Support Non-Power Test Reactor Design

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 9. Summary of Creep-Fatigue Tests to Support Non-Power Test Reactor Design

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
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Table 10. Summary of Potential Testing to Assess Stress Relaxation Cracking

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Non-Proprietary	Doc Number	Rev	Effective Date
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Table 11. Summary of Testing and Analysis Judged to be Warranted by the Materials PIRT Review

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
	KP-TR-013-NP-A	4	September 2022

Table 12. Overall Effects that will be Assessed to Develop Corrosion Rate Models

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 13. Detailed Plans for Corrosion Testing

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
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Table 14. Summary of Slow Strain Rate Testing to Assess Environmentally Assisted Cracking

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 15. Conditions for Corrosion Fatigue Crack Growth Rate and Stress Corrosion Cracking Tests

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 16. Test Conditions to Assess Creep-Rupture Performance in Flibe

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
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Table 17. Specimens for Characterization to Assess Metallurgical Effects

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Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 18. NOT USED

Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 19. NOT USED

Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor			
Non-Proprietary	Doc Number	Rev	Effective Date
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Table 20. Summary of Observations Intergranular Corrosion and EAC in Structural Alloys in Fluoride Salts or in Known Embrittling Contaminant

Alloys	Environment	Temp. (°C)	Time (hr)	Stress (MPa)	Notes	References
OFHC Cu	MSRE Fuel Salt (LiF-BeF ₂ -UF ₄)	>500	13,000	Unknown. Plastically Deformed	Extensive IGSCC, unclear if corrosion-deposit (rich in Ni, Fe, Cr, Mo) influenced.	Reference 9
INOR-8	11.2NaF-41KF-45.3LiF-2.5UF ₄ (mol %)	593-982	≤1200	14-207	Creep tests. Cracking is intergranular in both air and molten salt. Salt exposure does not degrade creep life.	Reference 97
INOR-8	Air with intentional contamination of Te, Se, S, etc.	~650	~2000	Strained to failure	S, Se, Te cause intergranular cracking. Cracks are deeper if stressed material exposed to Te	Reference 99
Ni Alloys	LiF-BeF ₂ -ThF ₄ 72-16-12 mol.%)	~645	Not specified	Not specified	Ni-S eutectic noted to lead to intergranular attack and failure of processing equipment	Reference 25
304 SS Hast. N, Hast. S, A600 Mod. Hast. N	Fuel salt (Flibe) contaminated with Cr ₃ Te ₄ and Cr ₅ Te ₆	700	2500	Post-exposure strain to failure at room T	Ni alloys are susceptible to Te induced IG cracking under oxidizing potentials. 304 SS & other Fe alloys resistant. ~1.5 wt.% Nb to Hast N imparts resistance, Ti+Nb negates effect.	Reference 20
Ni alloys near Hast N.	Fluoride Salts + Cr ₃ Te ₄	750	400	80 MPa	Applied stress increases the number density and depth of intergranular cracks	Reference 21
Ni-16Mo-6.4Cr-3.6Fe	FLiNaK	700	~250	190 MPa (~75% YS, 40% of UTS) (creep)	Extensive intergranular cracking. Creep failure time ~ 50% that in air. Salt thought to accelerate creep failure by preferentially corroding Mo depleted grain boundaries	Reference 98
GH3535 (Ni-16Cr-6Cr)	Te Vapor	25	150 hrs at 800°C prior to room temperature test	Tensile test to failure	≤90 μm of intergranular cracking.	Reference 101
Hastelloy N	FLiNaK + 500 ppm SO ₄ ²⁻	700	400	C-ring, 25% UTS (elastic) and 75% UTS (plastic)	Increase corrosion relative to no SO ₄ ²⁻ but no SCC. Stress increases depth of oxidized layer with sulfate but stress has little effect on Cr loss depth. S diffuses into grain boundaries and (Mn,Cr)S observed	Reference 108
316L/316	FLiNaK	600	≤125	1e-6 SSRT to failure	~1e-6 SSRT test to failure. Gauge shows extensive IG, shoulder less, grip none. Analysis indicates sulfur embrittlement under oxidizing conditions	This Work
316H	Flibe	600	≤125	1e-6 SSRT to failure	Limited grain boundary cracking observed. No association with sulfur. Tensile elongation ³ air testing.	

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Table 21. Comparison of the Expected Reactor Vessel Irradiation Damage

Reactor System	Expected Vessel DPA at End of Life	Expected Helium in Reactor Vessel at End of Life
Non-Power Test Reactor	<0.1	<10 at. ppm
Commercial Power Reactor		<15 at. ppm

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Figure 1. NOT USED

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Figure 2. Overview of the Commercial Power Generation Reactor Heat Transport Loops with Nominal Operating Temperatures

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Figure 3. Comparison of the Operating Pressures and Temperatures of Selected Conventional and Advanced Reactor Designs

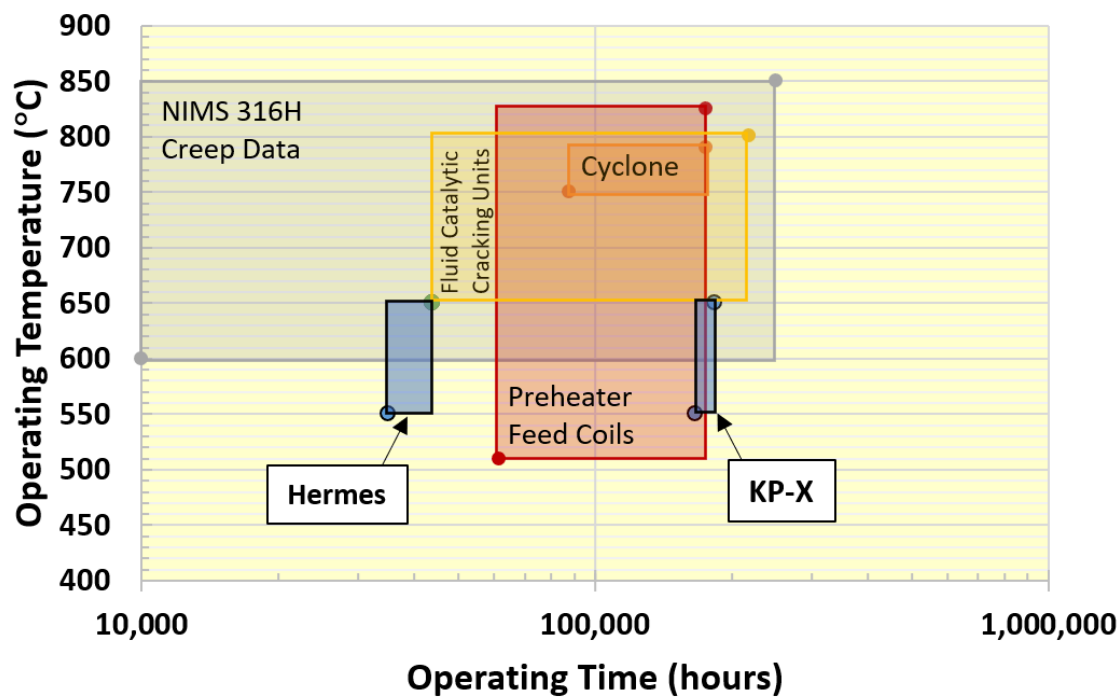
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Note: The labels refer to pressurized water reactors (PWR), boiling water reactors (BWR), high temperature gas reactors (HTGR), and sodium fast reactors (SFR)

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Figure 4. Comparison of the Operating Conditions of Alloy 316H in the KP-FHR (blue box) with Oil & Gas Refinery Components and Existing Creep Rupture Data



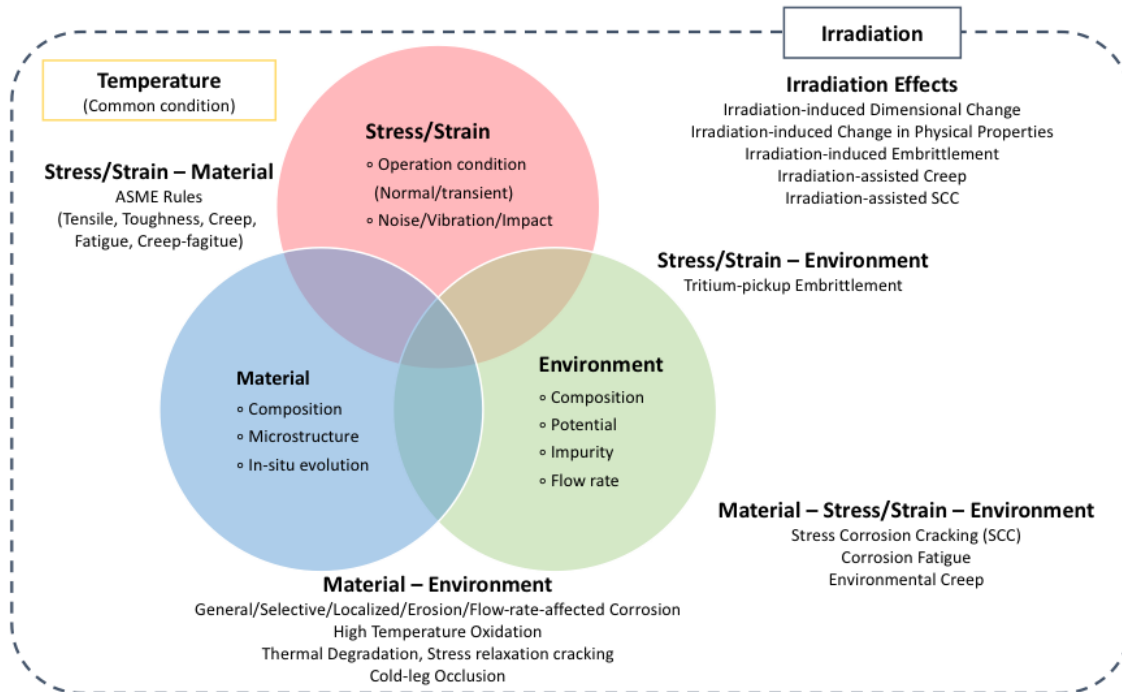
Note: Application of Alloy 316H and its weld metals in the KP-FHR is consistent with other industrial use

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Figure 5. NOT USED

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Figure 6. Illustration of the Environmental Degradation Mechanisms Considered in the Kairos Power PIRT Review of Environmental Degradation



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Figure 7. The Knowledge and Importance Rankings Used by the Expert Panel to Assess Environmental Degradation Phenomena

	Knowledge		
	<i>High</i>	<i>Medium</i>	<i>Low</i>
Importance <i>High</i> <i>Medium</i> <i>Low</i>	Category #4	Category #2	Category #1 (most important)
	Category #6	Category #5	Category #3
	Category #9 (least important)	Category #8	Category #7

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Figure 8. Summary of the PIRT Rankings

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Figure 9. Illustration of the Combined Corrosion Monitoring Approach

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Figure 10. Illustration of Slow Strain Rate Testing (SSRT) Data (left) and How the Results May Be Used to Map Out Regimes of Susceptibility to Environmentally Assisted Cracking (right)

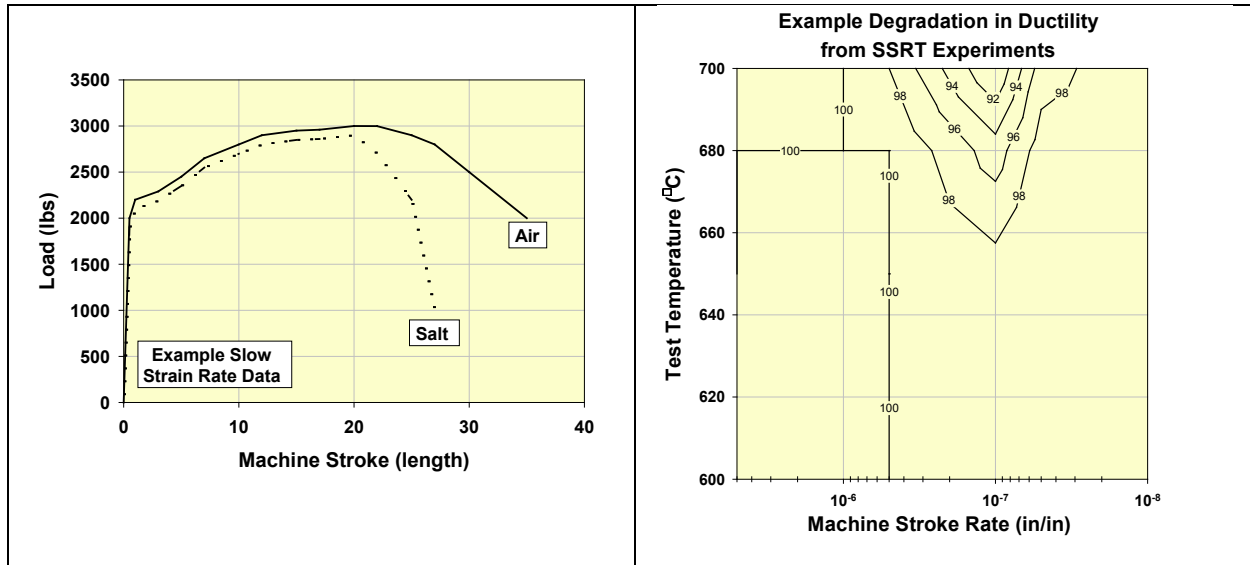


Figure 11. Example Corrosion Fatigue Crack Growth Rate Data (left) and How They Will be Compared to Data Collected in Air (right) to Assess the Effect of Environment

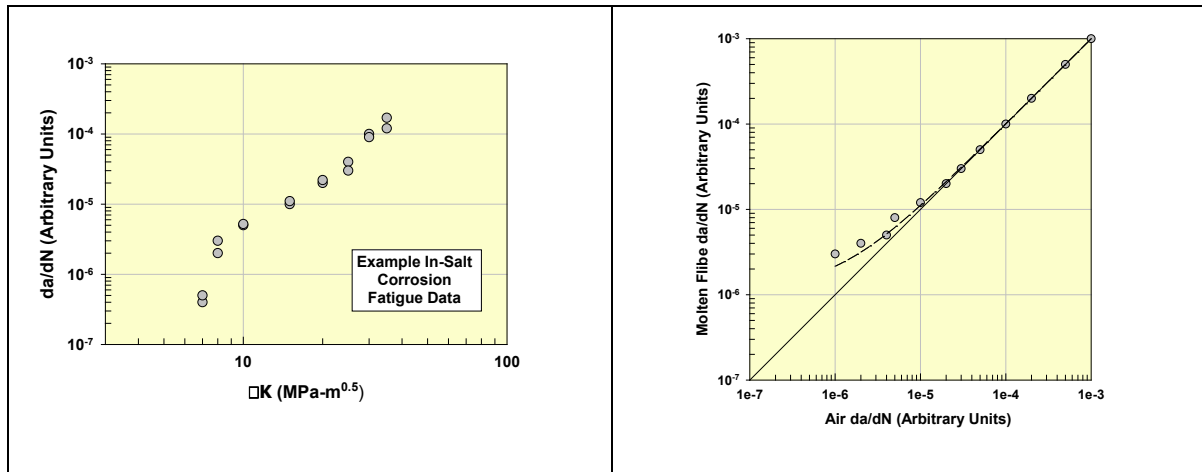
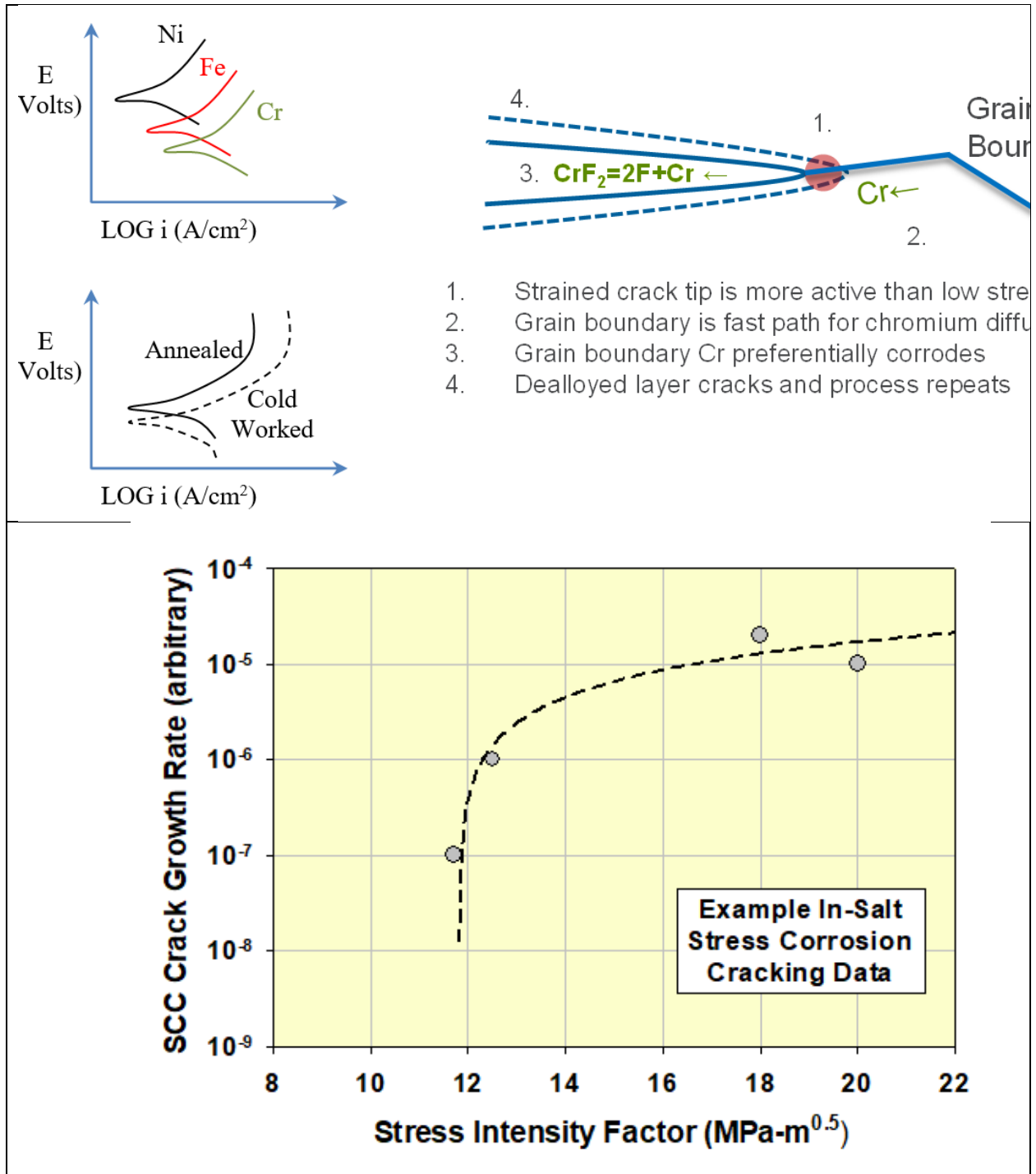
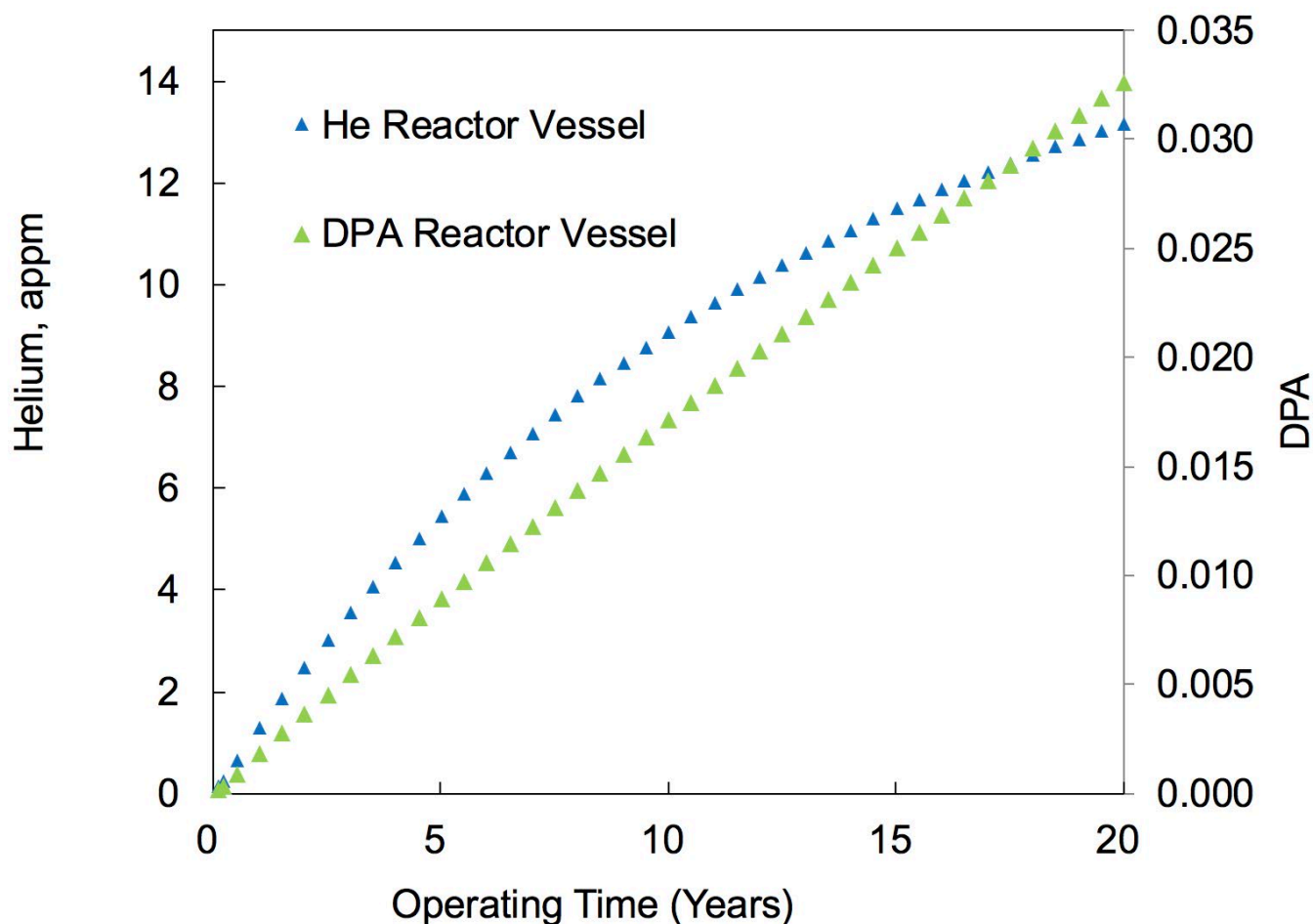


Figure 12. Illustration of a Potential SCC Mechanism in Flibe (top) where Grain Boundary Cr Loss is Accelerated at a Strained Crack Tip and (bottom) Schematic SCC Growth Rate Data



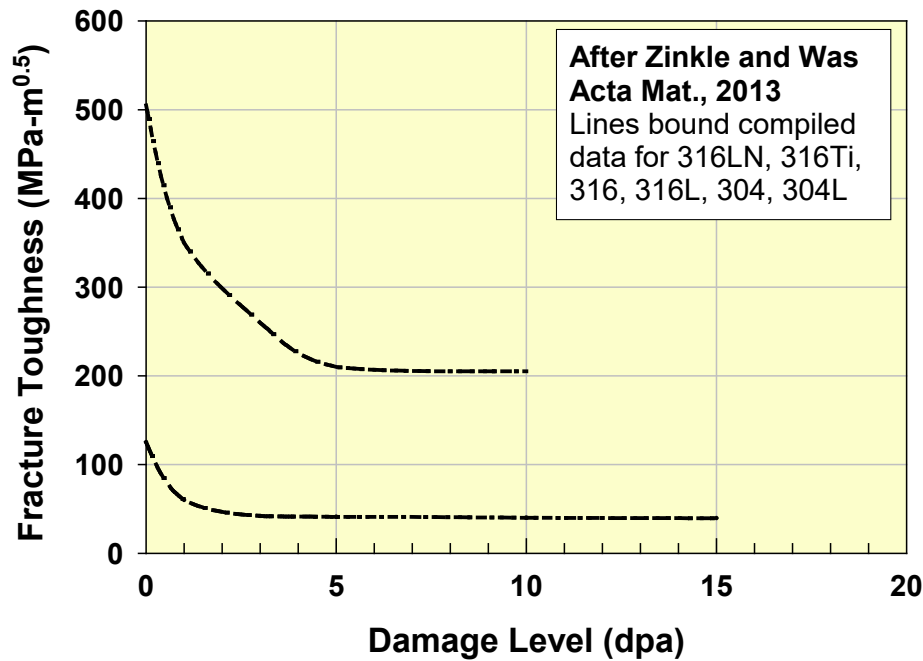
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Figure 13. Illustration of Irradiation Dose and Helium Production Vary as a Function of Time in the Commercial Power Reactor Vessel

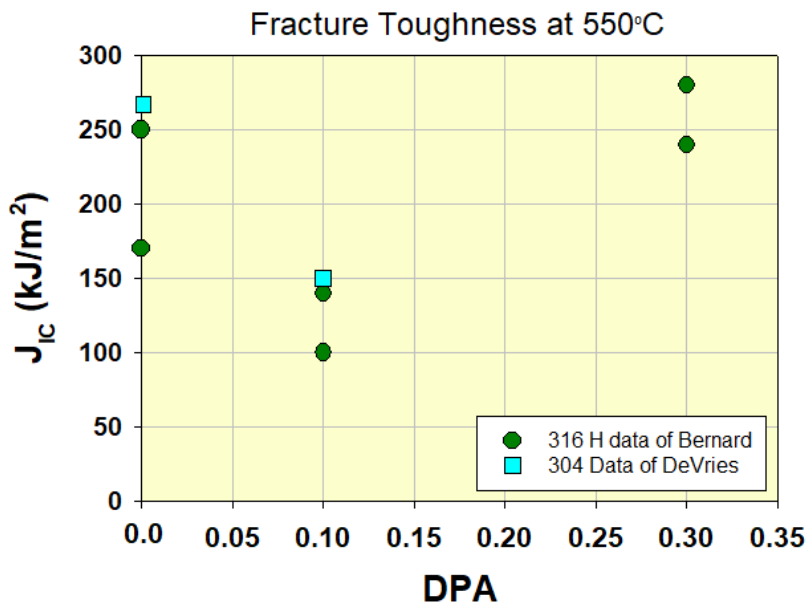


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Figure 14. How Irradiation Affects Fracture Toughness in Austenitic Stainless Steels and Specific Data for Alloy 316 and 304 at 550°C.



Note: Reference 52



Note: Reference 53 and 54

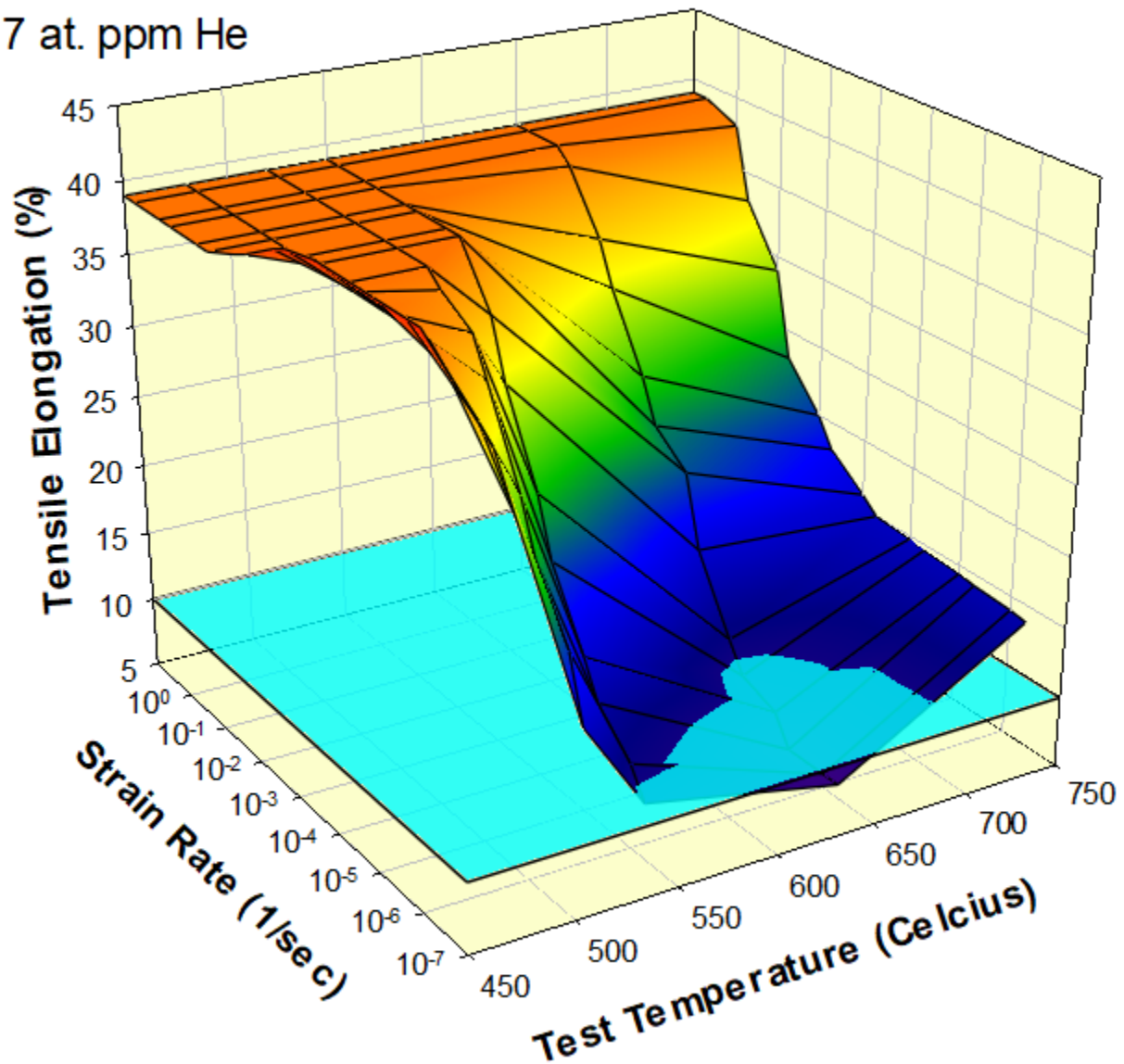
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Figure 15. Illustration of How Strain Rate and Temperature Affect Tensile Ductility in an Austenitic Stainless Steel Irradiated to a Helium Content of ~7 at. ppm

Data of De Vries 1979

Thermal Fluence: $2 \times 10^{24} \text{ n/m}^2$

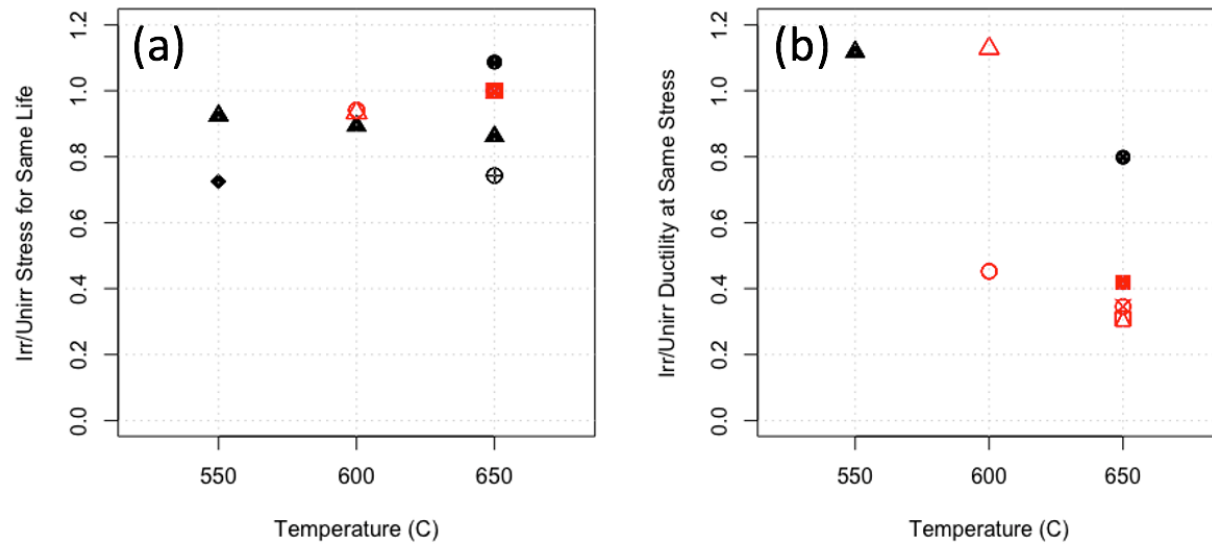
7 at. ppm He



Note: Reference 55

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Figure 16. (a) Normalized Creep Strength After Irradiation (Ratio of Irradiated Stress to Unirradiated stress to Reach the Same Average Creep Life) (b) Normalized Creep Ductility After Irradiation (Ratio of Irradiated Ductility to Unirradiated Ductility at the Same Stress)



- ▲ 316LN // 0.1dpa 2appm He // Aoto, K., et al. (1995)
- ◆ 316LN // 0.1dpa 3appm He // Aktaa, J. et al. (2002)
- ⊕ 316 // 0.05 dpa // Pfeil, P., & Harries, D. (1965)
- 316L // 0.05 dpa // Pfeil, P., & Harries, D. (1965)
- 17-8-2 // 0.25 dpa 2 appm He // Tavassoli (1996)
- △ 17-8-2 // 2 dpa 44 appm He // Tavassoli (1996)
- ⊗ 316 Weld // 3.6 dpa 2 appm He // Ward (1974)
- ⊠ 316 Weld 1065C, 1hr // 3.6 dpa 2 appm He // Ward (1974)
- 316 Weld 800C, 10h // 3.6 dpa 2 appm He // Ward (1974)

Note: References 56, 66, 67, 68, 69

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Figure 17. Variable Corrosion Rate of Alloy 316 Stainless Steel with Time (top) and The Strong Benefit of Be Addition (Redox Control) (bottom).

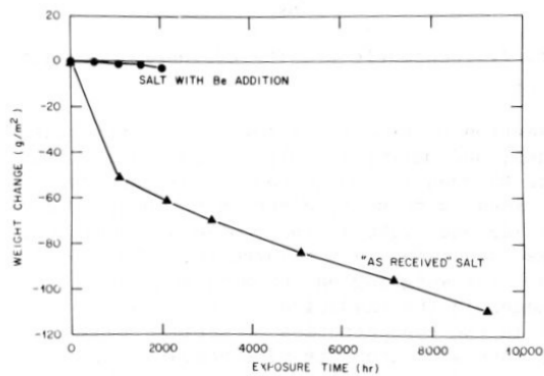
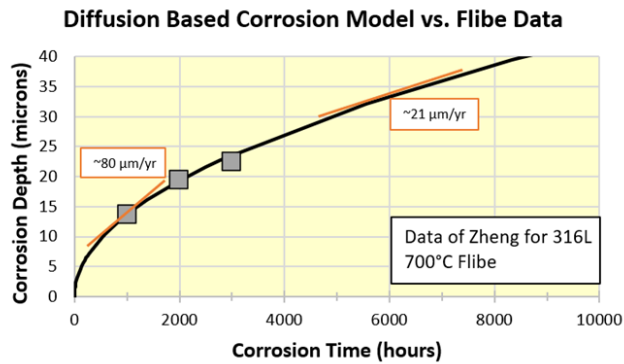
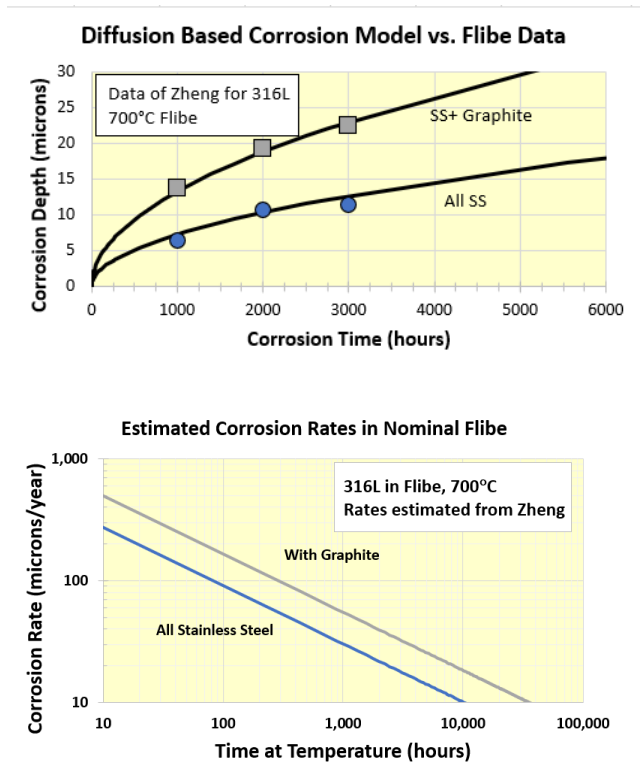


Fig. 2. Weight change versus exposure time for type 316 stainless steel in LiF-BeF₂ salt at the maximum loop temperature of 650°C.

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Figure 18. Data of Zheng et al, Illustrating the Effect of Graphite on the Corrosion Depth (top) and Corrosion Rate of Alloy 316L in Flibe at 700°C (right)



Note: The difference in corrosion rate is just less than 2X

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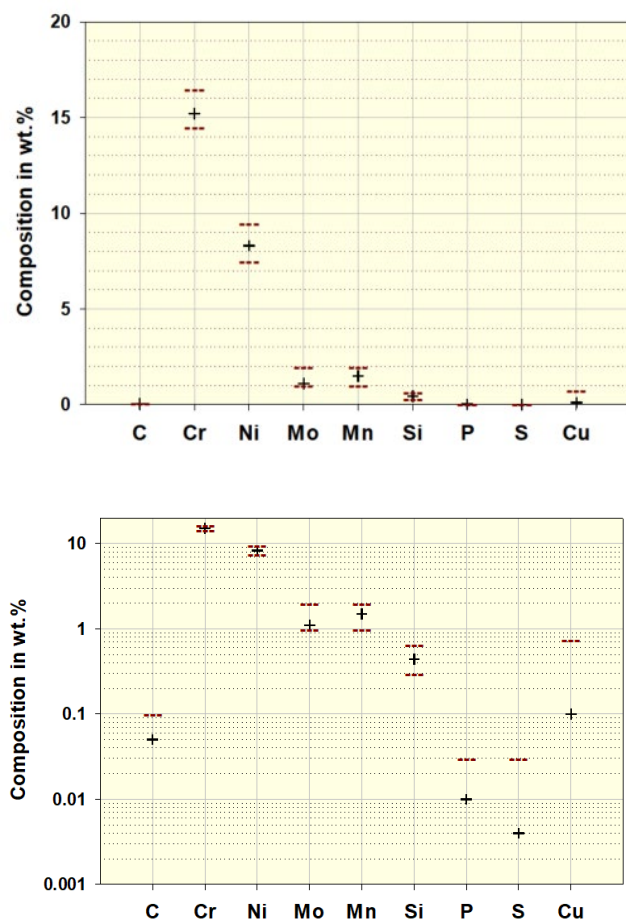
Figure 19. Examples of Weld Pad Buildups (top) and A V-Groove Weld (bottom) used to Fabricate Test Samples

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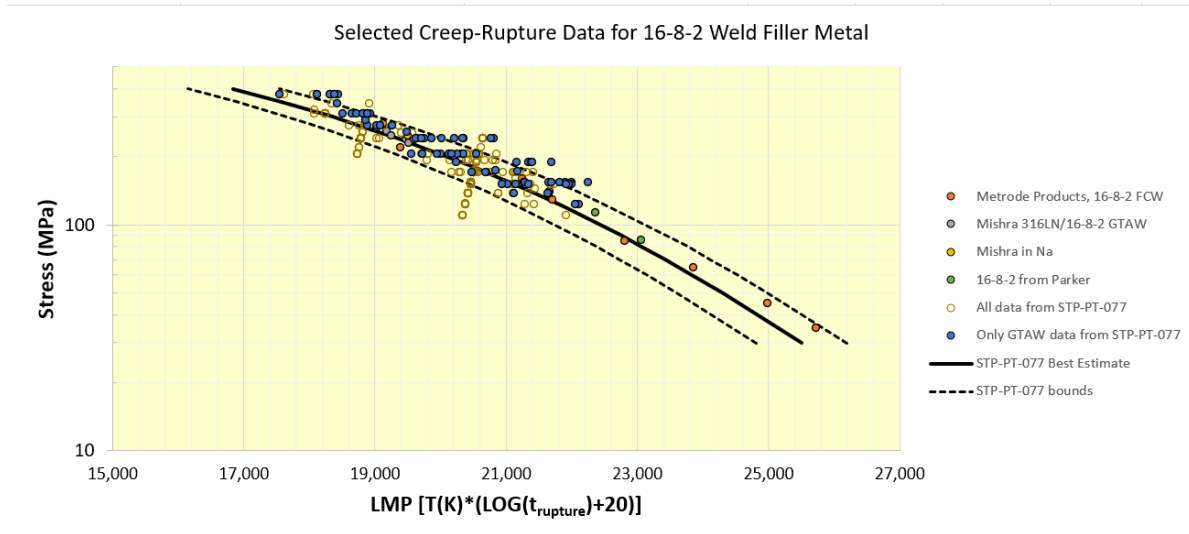
Figure 20. Comparison of the Composition of Heat 578409 (+ symbols) Relative to the ASME Code Specification (dashed lines, from Section II. 2017)



Note: C, P, S, and Cu only have maximum specifications. Linear scale (top) and log scale (bottom) to better illustrate the elements at low concentrations.

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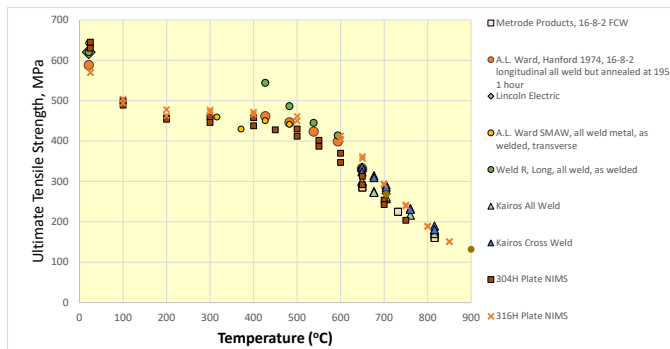
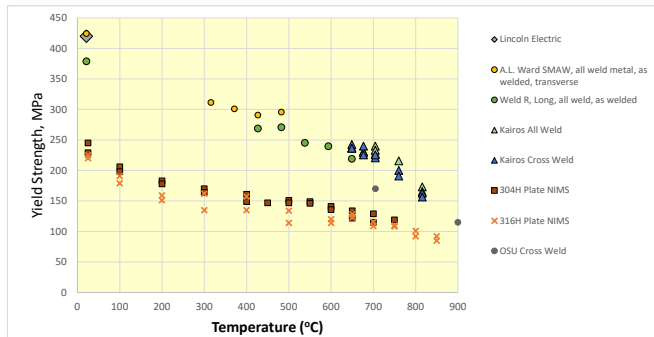
Figure 21. Selected Creep-Rupture Data for ER16-8-2 Weld Filler Metal Compared to the Best Estimate Prediction and Confidence Bounds



Note: Presented in STP-PT-077. The ASME Code Case data will be assessed relative to relevant data and appropriate statistical bounds determined.

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Figure 22. Comparison of Selected Base Metal and Weld Metal Tensile Data



Note: Yield strength shown on the top and ultimate tensile strength on the bottom.

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Figure 23. Weld Designs that Minimize the Risk of Stress Relaxation Cracking

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Figure 24. Illustration of a Narrow Groove Gas-Tungsten-Arc Weld with the Location of a Notched Tensile Bar Overlayed in the Heat Affected Zone

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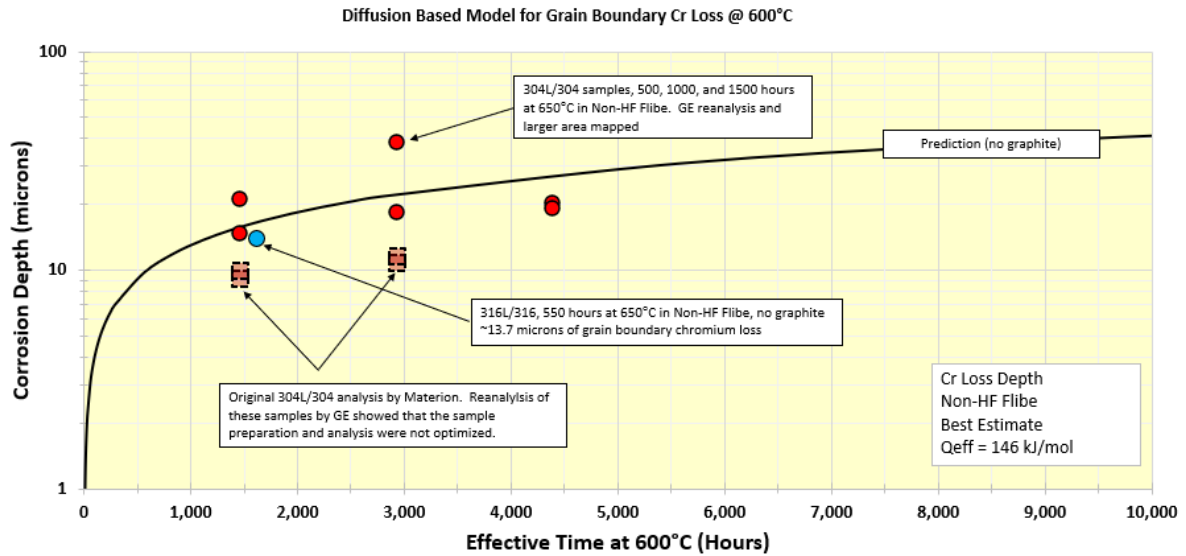
Figure 25. Potential Location of the CCS Relative to the Reactor Vessel

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Figure 26. Predicted Grain Boundary Diffusion Rate



Note: Predicted grain boundary diffusion rate (solid black line) with recent data for Alloy 304 (red points) and Alloy 316 (blue point). For the Alloy 304L/304 data, two separate analyses are included but the red points (large area, GE analysis) are judged to be the most accurate. Based on the small surface area of graphite in the Alloy 304L/304 tests, comparison with the no graphite curve is appropriate.

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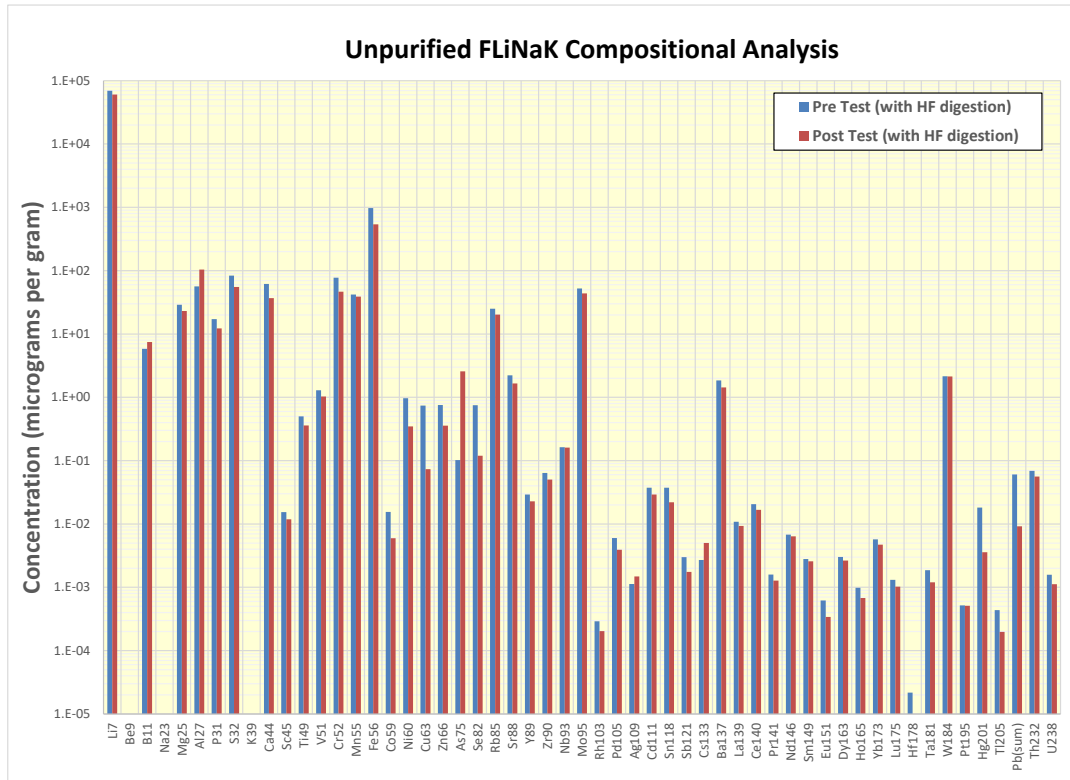
Figure 27. Schematic Illustration of a Rotating Cage Loop (RCL) Corrosion Testing System (left) and an Operational RCL System (right)

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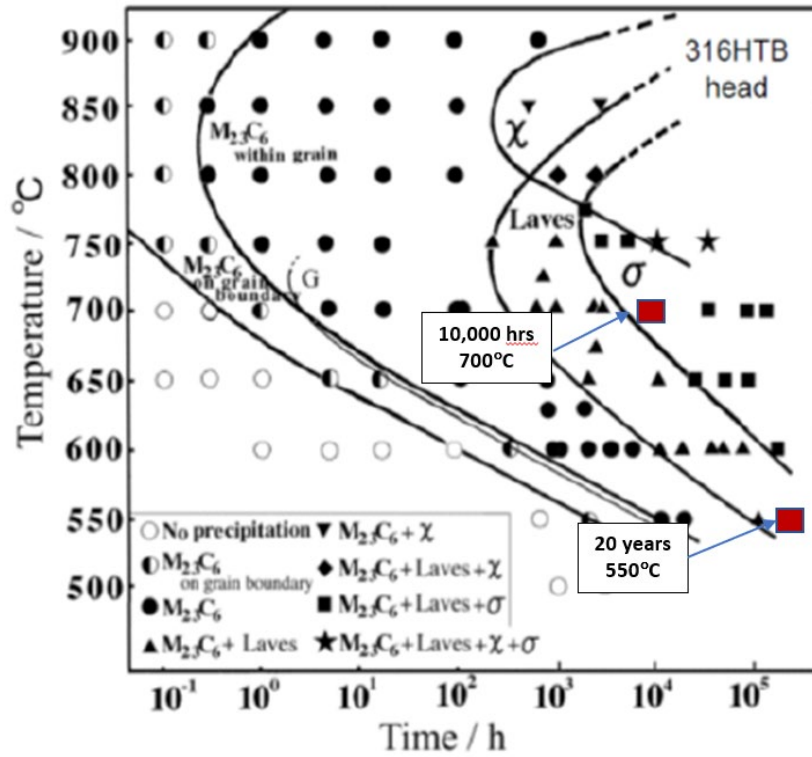
Figure 28. Example of Fluoride Salt Compositional Analysis



Note: Unpurified FLiNaK before and after SSRT testing an Alloy 316L sample at 600°C and 1e-6 (in/in)/sec

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Figure 29. The Time-Temperature-Transformation Diagram for Alloy 316H



Note: Targeting 700°C for 10,000 hours as an aging treatment to represent long time operation at 550°C (Reference 82)

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Figure 30. Calculated LiF-BeF₂ Phase Diagram Against Experimental Data

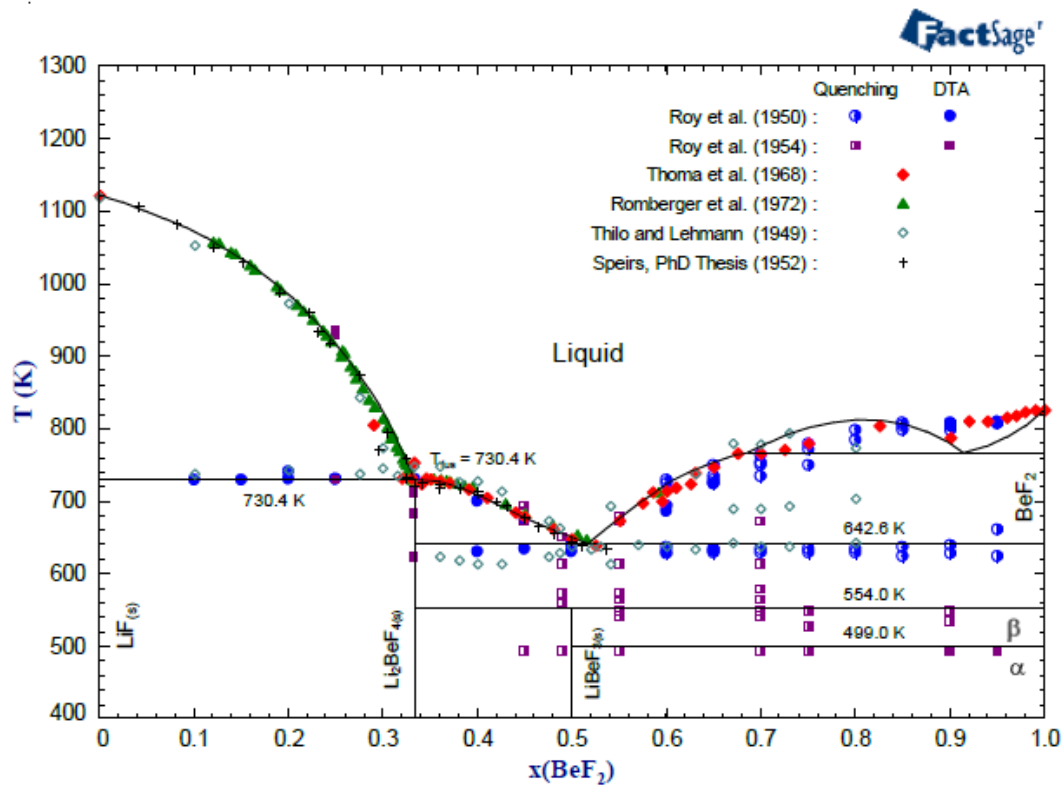
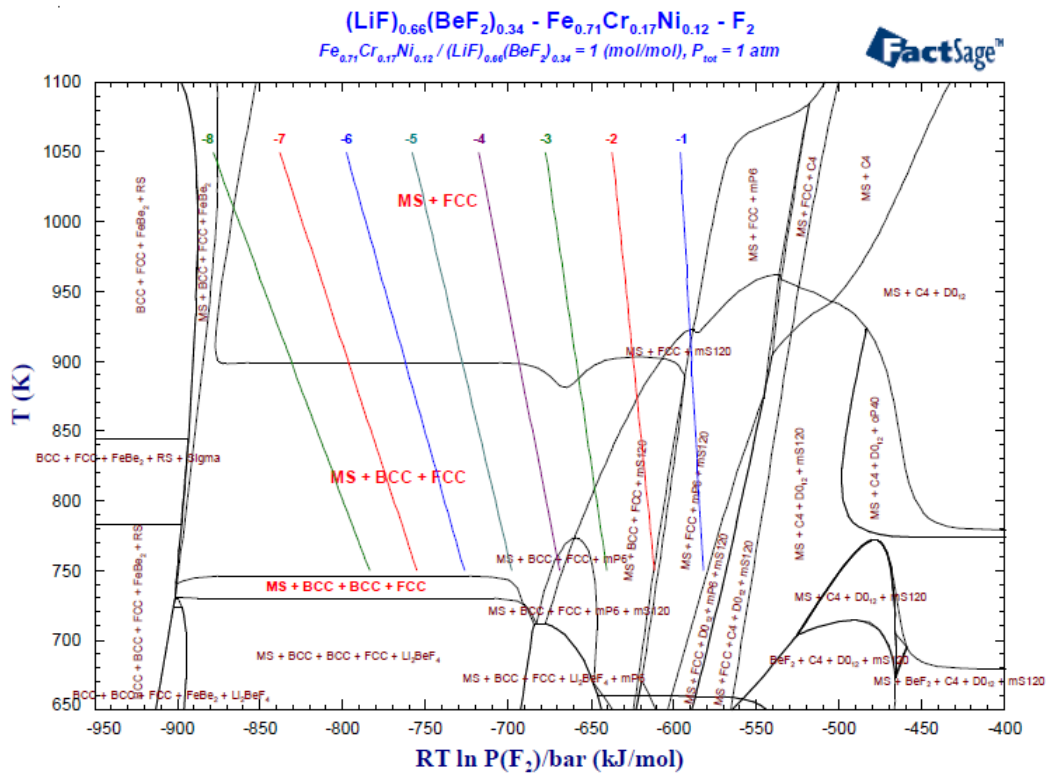


Figure VI-1 : Calculated LiF-BeF₂ phase diagram.

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Figure 31. Calculated Multicomponent Phase Diagram with Superimposed Log (p(HF))



Note: The diagram shows regions of metal stability as well as the concern of FeBe_2 formation if conditions are overly reducing.

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Figure 32. Schematic of the In-Situ Mechanical (ISM) Testing Systems (left) and an Operational ISM Running a Slow Strain Rate Test in FLiNaK Salt (right)

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Figure 33. Comparison of the 316 FLiNaK Fatigue Crack Growth Rate Data with Similar Test Conducted in Air

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APPENDIX A. COATINGS, CLADDING, AND TRITIUM MANAGEMENT

The design of the KP-FHR does not require the use of cladding or coatings. However, these materials may be desirable to optimize the performance of the reactor system. [[

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Cladding and Coatings

Current ASME Section III Division 5 Code rules for clad structural components in elevated temperature service are limited. ASME Section III Division 5, Paragraph HBB-2121 allows non-Code qualified materials to be used for cladding if the clad thickness is 10% or less of the thickness of the base material. ASME Section III Division 5, Paragraph HBB-3227.8 specifies that no structural strength will be attributed to the cladding in satisfying the primary load stress limits. It also requires that the cladding will be considered in design evaluation related to limits on deformation-controlled quantities, i.e., strain accumulations due to ratcheting and creep-fatigue damage but does not provide guidance or requirements for that assessment. While the 10% clad thickness rule allows Kairos Power to select corrosion-resistant materials that are not Code qualified for Class A service, the lack of design rules presents challenges in their application.

In order to help enable the application of corrosion-resistant coatings and cladding, Kairos Power is has completed a GAIN research collaboration (References 70, 71, 72, and 73) (Gain cladding project under contract No. DE-AC02-06CH11357 with the US Department of Energy). The GAIN research includes establishing the mechanical nature of the cladding or coating (compliant or elastic), determining key mechanical properties (yield strength, creep rate), assessing the integrity of the coating after thermal cycling, and testing the environmental compatibility of the cladding or coating in molten Flibe salt. This program is expected to result in the ability to use cladding and coatings with ASME Section III Division 5 structural materials.

Coatings and claddings used in the KP-FHR to decrease tritium permeability will potentially be applied to salt facing surfaces because the benefit of a tritium permeation barrier is expected to increase for salt facing applications compared to the air side. Since the tritium permeation barrier coatings and claddings will be exposed to molten Flibe, the corrosion resistance of barrier materials will be evaluated, and the selection of tritium permeation barrier materials will be limited to those which provide comparable or improved corrosion rates in Flibe compared to Alloy 316H. In addition to Flibe facing tritium permeation barriers, the KP-FHR design may also include nitrate salt-facing coatings which would assist in reducing the permeation of tritium through the intermediate loop piping.

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Cladding and Coatings for Air Side Applications

For the application of cladding or coatings on the air side of safety-related systems, there are no ASME design rules governing their use. Potential applications for coatings on the air side of these systems include oxide films [[
]] surface treatments to enhance thermal emissivity or providing anti-galling. If coatings are used on the air side of safety-related systems, the benefit of the treatment will be demonstrated and confirmed through analysis and/or testing that there is no significant degradation to the underlying structural material.

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APPENDIX B. INSPECTION AND AGING MANAGEMENT

Introduction

Nuclear Power Plant component life management requires a combination of analysis, inspection, testing, and monitoring activities. The information derived from each of these activities complements one another and should be utilized as part of an integrated program. Qualification through mechanical and environmental testing is the first step in ensuring material performance for long-term service in nuclear power plants. While test plans can to some extent account for combinations of mechanical and environmental factors that affect material performance, it is rare that laboratory testing can account for all of the variables and interactions present during reactor operation. Furthermore, it is often impractical to perform laboratory tests for times on the order of the expected component lifetimes (usually decades). While the material qualification test programs described in this document provide confidence that Alloy 316H / ER16-8-2 will perform satisfactorily over the service life of the plant; in-service monitoring and evaluation throughout the plant life will be used to further ensure the safe and reliable operation of the KP-FHR.

Inspection and Monitoring Program

ASME Section XI has historically provided rules for in-service inspection and the replacement and repair of components during the operating life of light water reactors. The unique physical features of high temperature reactors such as the KP-FHR present a new paradigm for the Reliability and Integrity Management (RIM) that has required the Code to develop new approaches. The new approach being implemented as ASME Section XI Division 2 “Reliability”, applies to any type of reactor design and was published for the first time in the 2019 Edition of the ASME code.

For both the non-power test reactor and commercial power reactor systems a materials surveillance system will be used to assess the combined effects of molten salt and irradiation on structural materials. The location of the materials surveillance system for the non-power test reactor is planned to be outside of the core and within the graphite reflector. At the location of the metallic samples, the samples should see a flux of 0.02-0.03 dpa/year so that in a relatively short exposure time (~1 year exposure) a significant degree of irradiation could be produced and assessed via subsequent post-irradiation examination. The bounding dpa for the reactor vessel is <0.1 dpa at the end of life, so that 0.02 dpa represents a meaningful degree of damage.

In the materials surveillance system program, representative weld metal, base metal, and heat affected zone samples will be assessed for the potential occurrence of irradiation affected corrosion and stress corrosion cracking. For Irradiation-Assisted Stress Corrosion Cracking, fracture mechanics samples will be used that approximate the conditions of the in-Flibe testing, i.e., $K_I \sim 25 \text{ MPa}\sqrt{\text{m}}$ at an average temperature between 550-650°C. One difference is that these samples will be loaded via constant displacement rather than tested at constant K_I . The details of the surveillance coupon program (sample types, numbers, etc.) will be provided at the time of the operating license application.

For both the non-power test reactor and the commercial power reactor systems, inspection and monitoring programs to further ensure material and component performance will be utilized. These efforts will involve on-line monitoring systems as well as periodic inspections. A number of technologies

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are currently being assessed and developed to facilitate these programs that are applicable to the non-destructive inspection of the reactor vessel. For the non-power test reactor, the details of the inspection and monitoring programs will be provided at the time of the operating license application.

For the commercial power reactor, inspection and monitoring will be performed in accordance with the requirements of ASME Section XI 2019 Edition, Mandatory Appendix I (“RIM Decision flowcharts for use with the RIM”) and Appendix II (“Derivation of component reliability targets from plant safety requirements”). Component Level Reliability Requirements will be derived from Plant Level Reliability Requirements through the Probabilistic Risk Assessment process. With Reliability Targets established, components will be assessed for mechanisms of environmental degradation and modes of failure as derived from the Phenomenon Identification and Ranking Table. Critical flaw size will be determined for the most likely modes of failure in each component. Monitoring and Non-Destructive Examination technologies will be evaluated for the capability to detect sub-critical flaws and to endure the relevant inspection environments. Technologies and inspection schedules will then be selected for each area of interest to ensure that flaws can be detected before they grow to critical flaw size. Material performance will be monitored during operation, and data will be fed back to update the Monitoring and Non-Destructive Examination (MANDE) schedule throughout the life of the plant. The specific details of a RIM program for the commercial power reactor will be provided with the operating license application.

The new RIM allows a combination of MANDE methods for an aging management program. The ability to use both monitoring and non-destructive examination is a significant advantage to many advanced reactor designs, including the KP-FHR, since their compact size and need for coolant chemistry control limits access to some components during the operating lifetime of the plant. While the 2019 Edition of ASME Section XI Division 2 outlines the top-level requirements for a RIM program, Mandatory Appendix VII of Division 2 will describe the specific MANDE methods and acceptance criteria for each of the different types of advanced reactors. Note that Article VII-4 has been reserved for Molten Salt Reactors (and presumably FHR designs) but has not yet been developed in detail. Kairos Power is active with the Section XI Committee Sub-Groups and Working-Groups related to RIM and MANDE and plans to apply the KP-FHR experience to the development of relevant Code articles for FHRs.

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APPENDIX C. DETAILS OF THE CORROSION DATA ANALYSIS

Many testing programs that are expected to yield quantitative results were developed with the intent of statistical analysis of the data. For example, the general corrosion testing of Alloy 316H and ER16-8-2 plans include three samples per condition, conducted over a wide range of times and temperatures. These data will be analyzed via electron microscopy of corrosion coupon cross sections which, we believe, is superior and more sensitive a measure than weight change.

With those corrosion data, Kairos Power will develop ‘baseline’ corrosion models for Alloy 316H and ER16-8-2 and conduct separate effects tests to assess key variables that include microstructure, contaminants, redox control, occluded geometry, and erosion-corrosion. Statistical analysis such as Analysis of Variance (ANOVA) will be used to establish the significance of these variables on the response model of corrosion rate compared to random error. Furthermore, the corrosion model will utilize appropriate prediction bands to ensure appropriate and conservative extrapolation from test conditions to KP-FHR operational times and temperatures.

Note that some testing may not be amenable to statistical analysis but is being performed to develop understanding and guidance. For example, the slow strain rate tests in Flibe are being performed primarily to assess regimes in which environmentally assisted cracking may occur. In these tests, a change in response (load vs. stroke) relative to air testing and post-test analysis of the fracture path will be used to develop understanding of the degree of concern for cracking. Similarly, stress corrosion cracking tests are being conducted to better understand if this phenomenon occurs in environments and mechanical conditions of relevance to the KP-FHR. Depending on the response of Alloy 316H and ER16-8-2 to these tests, a statistical analysis of the data may be used but also may employ fundamental materials science and engineering judgement to develop appropriate design factors or other practices (e.g., periodic inspection) that will appropriately address the concern of environmentally assisted cracking.

Note that corrosion rates can be confounded by complicating factors such as carbon pickup during testing as well as the difficulty in removing dried salt from test coupons. To mitigate these factors, Kairos Power will use electron microscopy of corrosion coupon cross sections as the primary method to assess corrosion (e.g., depth of chromium loss) as well as other compositional changes (e.g., the extent of Fe and Ni loss, the precipitation of Mo-rich Laves phase, and the precipitation of carbon rich phases). An example of this analysis is given below in Appendix C, Figure 1.

For information, an example of expected statistical analysis of corrosion data is presented below. In this example, the corrosion data of Zheng et al. (pink squares) are used to generate example corrosion data for three different temperatures and for times up to 10,000 hours (Reference 18). The example data are shown below in Appendix C, Figure 2.

An example of how these data may be fit is via Appendix C, Equation C-1. In Equation C-1, A is a fitting constant, t is the exposure time, n is a fitting constant (equal to 0.5 for mass diffusion control), Q is the apparent activation energy, R is the gas constant and T the temperature.

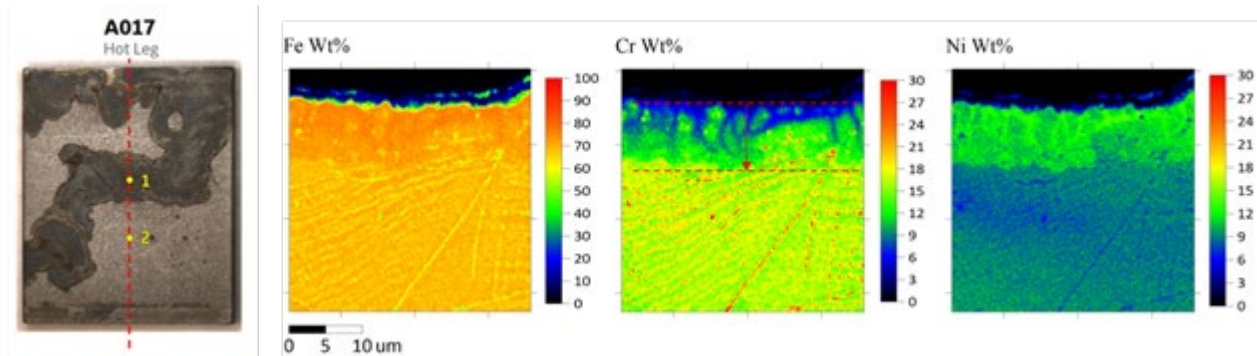
$$Cr \text{ loss depth} = A * t^n * EXP(-Q/RT) \quad \text{Eq. C-1}$$

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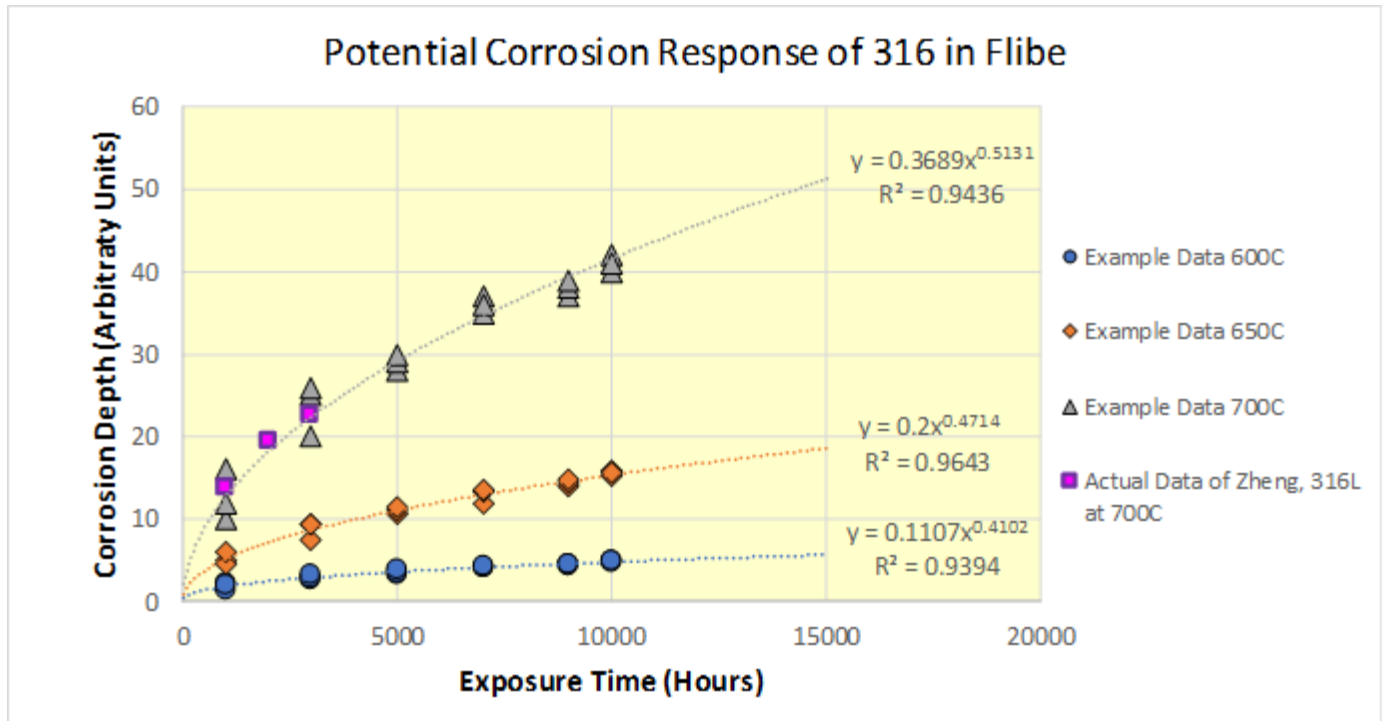
The fit of the data to the combined model is shown by the blue surface in Appendix C, Figure 3. In this manner all the data (example data in gray circles, Zheng data in pink squares) can be used to increase the confidence in extrapolating to the conditions of the KP-FHR. For example, the reactor vessel will operate at approximately 550°C for [[]] which would exhibit <10 microns of corrosion (Cr loss) via the best estimate prediction of this model. The example baseline model is shown in two dimensions in Appendix C, Figure 4 (upper graph) with 95% prediction intervals. In Appendix C, Figure 4, an example of a separate effects corrosion test is shown (lower plot) along with how a factor of improvement may be defined. In this example, the factor of improvement is conservatively determined at an exposure time within the data and between the baseline model lower bound and the separate effects test upper bound.

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Appendix C Figure 1. Example of How a Corrosion Coupon was Sectioned (left) and Corresponding Compositional Maps for Iron, Chromium, and Nickel



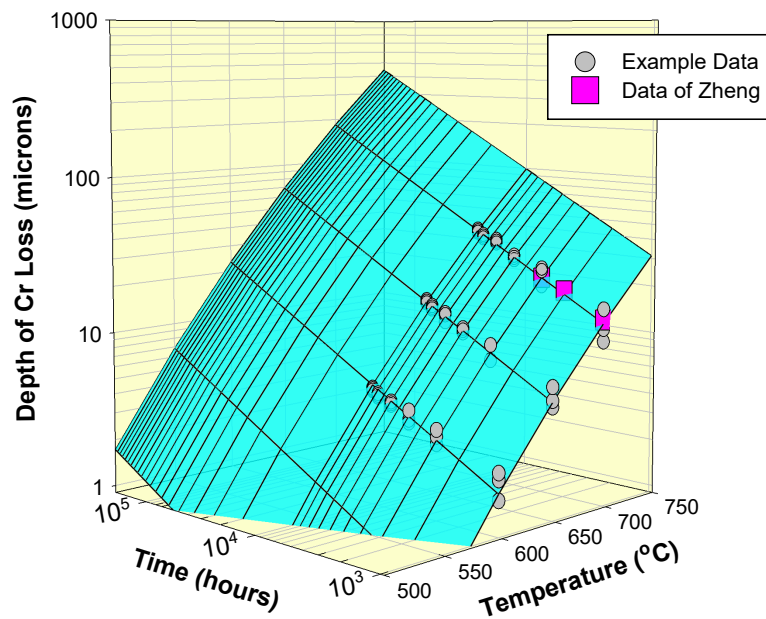
Appendix C Figure 2. The Corrosion Data of Alloy 316L in Flibe of Zheng (Pink Squares) Compared to Example Data at Three Different Temperatures



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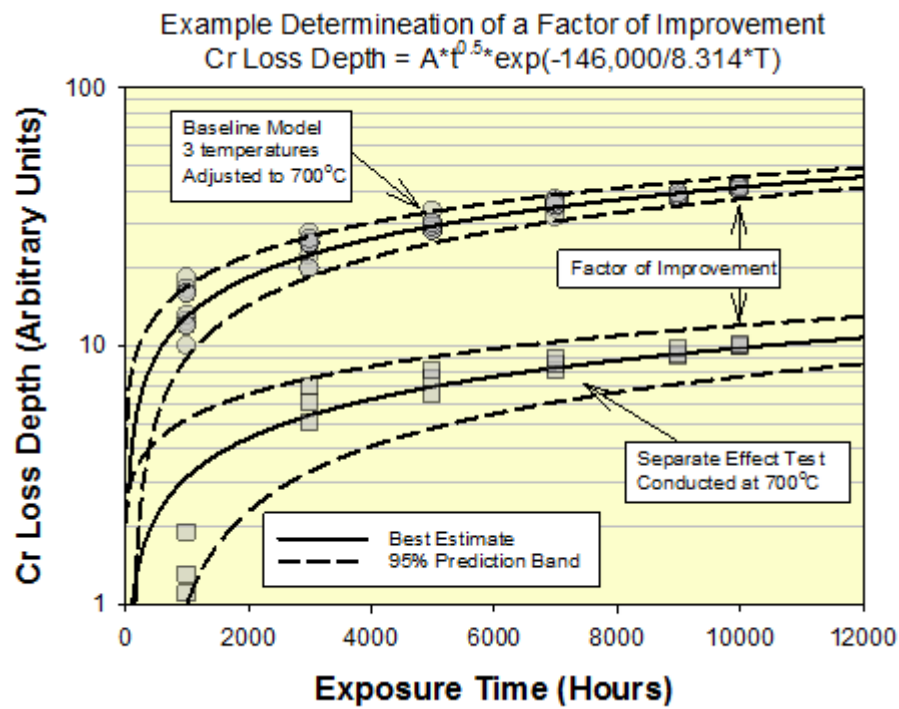
Appendix C Figure 3. Example of How Corrosion Data May be Fit and Extrapolated to Times Out to 20 years

Example Fitting and Extrapolation of Corrosion Data for Development of 'Baseline' Corrosion Response



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Appendix C Figure 4. Example of How the Baseline Corrosion Model May be Compared to a Separate Effects Test to Determine a Factor of Improvement



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APPENDIX D. NOT USED

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APPENDIX E. NOT USED

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APPENDIX F. CERTIFIED MATERIAL REPORTS

Appendix F Figure 1. Material Certification Report for Alloy 316H Plate

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Appendix F Figure 2. Overcheck of the Composition of the Alloy 316H Plate

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Appendix F Figure 3. Material Certification Report for the ER16-8-2 Weld Wire

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Appendix F Figure 4. Material Certification Report for Second Heat of ER16-8-2

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Appendix F Figure 5. Material Certification Report (tentative) for a Third Heat of ER16-8-2

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