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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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KAIROS POWER LICENSING SUBCOMMITTEE

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OPEN SESSION

+ + + + +

THURSDAY

JANUARY 12, 2023

+ + + + +

The Subcommittee met, via Teleconference,
at 9:30 a.m. EST, David A. Petti and Ronald G.
Ballinger, Chairs, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chair

RONALD G. BALLINGER, Chair

VICKI M. BIER, Member

CHARLES H. BROWN, JR., Member

VESNA B. DIMITRIJEVIC, Member

WALTER L. KIRCHNER, Member

GREGORY H. HALNON, Member

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JOSE MARCH-LEUBA, Member

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

DENNIS BLEY

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIALS:

WEIDONG WANG

CHRISTOPHER BROWN

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Kairos Power Topical Report (KP-TR-013), "Metallic
Materials Qualification for the Kairos Power Fluoride
Salt-Cooled High-Temperature Reactor"

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MEMBER BALLINGER: Okay. It's now a little after 2:00. The meeting will come to order.

This is a meeting of the Kairos Power Licensing Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Ron Ballinger, technologically illiterate Chairman of today's Subcommittee meeting.

ACRS members in attendance are Charles Brown, Joy Rempe, Matt Sunseri -- keep going down the line here -- Vicki Bier, Dave Petti, Greg Halnon, Jose March-Leuba, and our Consultants Dennis Bley and Steve Schultz are also participating. And if I missed somebody, I apologize.

MEMBER KIRCHNER: Ron, Walt, I'm here.

MEMBER BALLINGER: Ah, okay, Walt.

MEMBER DIMITRIJEVIC: I am here, too.

MEMBER BALLINGER: And Vesna. Boy, I've

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1 got two strikes on me already.

2 Okay. Chris Brown of the ACRS staff is
3 the Designated Federal Official for this meeting.

4 During today's meeting the Subcommittee
5 will review staff's Safety Evaluation on Topical
6 Report "Metallic Materials Qualification for the
7 Kairos Power Fluoride Salt-Cooled High-Temperature
8 Reactor," Revision 4.

9 The Subcommittee will hear presentations
10 by and hold discussions with the NRC staff, Kairos
11 Power representatives, and other interested persons
12 regarding this matter.

13 Part of the presentations by the Applicant
14 and the NRC staff may be closed in order to discuss
15 information that is proprietary to the licensee and
16 its contractors, pursuant to 5 USC 552b(c)(4).
17 Attendance at the meeting that deals with such
18 information will be limited to NRC and its
19 consultants, Kairos Power, and those individuals and
20 organizations who have entered into an appropriate
21 confidentiality agreement with them. Consequently, we
22 need to confirm that we have -- at that time, if we
23 need a closed session, we'll need to confirm that we
24 have only authorized people available, and we'll have
25 a different invitation for this.

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1 The rules for participation in all ACRS
2 meetings, including today's, were announced in The
3 Federal Register on June the 13th, 2019.

4 The ACRS section of the U.S. NRC public
5 website provides our Charter, Bylaws, agendas, Letter
6 Reports, and full transcripts of all full and
7 subcommittee meetings, including slides presented
8 here. The meeting notice and agenda for this meeting
9 were posted there.

10 We have received no written statements or
11 requests to make an oral statement from the public.

12 The Subcommittee will gather information,
13 analyze relevant issues and facts, and formulate
14 proposed positions and actions, as appropriate, for
15 deliberation by the full Committee.

16 The rules for participation in today's
17 meeting have been announced as part of the notice of
18 this meeting previously published in The Federal
19 Register.

20 A transcript of the meeting is being kept
21 and will be available, as stated in The Federal
22 Register notice.

23 Due to the COVID pandemic, today's meeting
24 is being held over Microsoft Teams for the ACRS, NRC
25 staff, and the licensee's attendees. There is also a

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1 telephone bridge line allowing participation of the
2 public over the phone.

3 When addressing the Subcommittee, the
4 participants should, first, identify themselves and
5 speak with sufficient clarity and volume, so that they
6 may be readily heard. When not speaking, we request
7 that participants should mute your computer
8 microphone, or phone by pressing *6, if you've had it
9 on.

10 I'll now proceed with the meeting, and I'd
11 like to call on the NRC staff. I'm not sure who is
12 going to make a comment for the NRC. If there's a
13 member that wants to make a comment, please do so.

14 MR. RIVERA: Good afternoon. This is
15 Richard Rivera, Quality Manager for the review of this
16 Topical Report.

17 The remarks that were covered by my Branch
18 Chief, William Jessup, that were made this morning,
19 also cover the discussion of this Topical Report. So,
20 we have no additional remarks.

21 MEMBER BALLINGER: Thank you.

22 MR. RIVERA: Thank you.

23 Okay. So, first up -- did I hear another
24 person that might want to make a comment?

25 (No response.)

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1 Okay. Thank you.

2 Okay. So, Kairos, I guess it's George
3 Young. Are you up?

4 MR. PRICE: Hello. This is John Price.
5 Can you hear me okay?

6 MEMBER BALLINGER: Ah, sorry about that.
7 Yes, I can.

8 MR. PRICE: Okay. Okay. We'll go ahead
9 and get started.

10 Hello. My name is John Price. I'm a
11 Senior Licensing Engineer for Kairos Power, and I'll
12 be presenting with Dr. George Young, on the Metallic
13 Materials Qualification Licensing Topical Report,
14 Revision 4.

15 I have a bachelor's degree in nuclear
16 engineering from Texas A&M University, and I'm proud
17 to say that I've been in the regulatory part of this
18 industry for over 40 years, in advanced reactor and
19 SMR design for the past 10 years, and with Kairos
20 Power for the past four years.

21 As we always start off, our company's
22 vision is to enable the world's transition to clean
23 energy, with the ultimate goal of dramatically
24 improving people's quality of life while protecting
25 the environment. What this means as we go through

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1 this meeting is that, as the Licensing Topical Report
2 gets approval from the ACRS, we are one step closer to
3 transitioning to cleaner energy and improving people's
4 qualify of life while protecting the environment.

5 Today, Dr. Young and I would like to
6 present the Metallic Materials Qualification Topical
7 Report and our methods used to qualify these materials
8 for use in the KP-FHR, specifically addressing the
9 environmental effects on materials.

10 The testing plan is for metallic materials
11 used in Flibe-wetted areas for safety-related, high-
12 temperature components in the non-power test reactor
13 which we call Hermes, and for the commercial power
14 reactor which we'll designate as the KPX reactor.

15 The Topical Report uses a phenomena
16 identification and ranking table, the PIRT test plan,
17 as a baseline for the KP-FHR -- this is from Reg Guide
18 1.203 -- used to identify significant degradation
19 phenomena and to develop the testing and modeling
20 qualification presented in the Topical Report. As
21 this is a methodology document, the demonstration and
22 qualification will be documented in the Safety
23 Analysis Reports as part of our future licensing
24 action.

25 Any questions on this?

1 MEMBER KIRCHNER: Yes. Hi. This is Walt
2 Kirchner.

3 Just a general question. You talked about
4 the Flibe-wetted areas of the primary system. Are you
5 considering other materials for internals, and you
6 would address those in a separate TR? Or is your
7 intent, by and large, to use the two materials
8 identified here as internals for the reactor vessel
9 and primary coolant system?

10 MR. PRICE: Yes. So, the 316H materials
11 will be used for the safety-related areas for the
12 Hermes and the KPX. We may get into using other
13 materials for non-safety-related areas. But, for the
14 safety-related areas, these are the materials that
15 we'd be using there, covered by this report.

16 Does that answer your question?

17 MEMBER KIRCHNER: Yes, it does. Thank you
18 very much. I was just curious whether you were
19 considering Inconels or other materials for things
20 like control rods and such. Thank you.

21 MR. PRICE: So, this slide right here is
22 a depiction of table 1 from the Metallic Materials
23 Topical Reports and provides a summary of key
24 parameters for the commercial power reactor, as well
25 as the non-power test reactor.

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1 The main thing to note on here is the
2 lifetime of the non-power test reactor will only be
3 one year commissioning and four years of operation.
4 For this reason, as we'll discuss later on, there will
5 be a reduced test duration for the shorter-lived non-
6 power test reactor.

7 Slide 5 represents the organization of the
8 Licensing Topical Report. From this, I'd like to say
9 that this test plan and analysis ensures extremely low
10 probability of abnormal leakage, rapidly propagating
11 failure, or gross rupture, which partly satisfies the
12 principal design criteria of PDC 14, the Reactor
13 Coolant Boundary, and PDC 31, which is the Fracture
14 Prevention of the Reactor Coolant Boundary.

15 At this time, I'm going to turn it over to
16 Dr. George Young to complete the open session
17 presentation. Dr. Young is a fellow scientist at
18 Kairos Power and has a BS in materials engineering
19 from Renssalaer Polytechnic Institute and an MS and
20 PhD degrees in material science from the University of
21 Virginia. He has over 30 years of experience in the
22 nuclear power industry and is expert with material
23 selection and performance for both conventional and
24 advanced nuclear power systems.

25 At Kairos Power, Dr. Young leads the

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1 structural material qualification effort and
2 environmental degradation testing. Dr. Young has
3 authored over 50 peer-reviewed articles and
4 books/chapters in the research area of
5 environmentally-assisted cracking, weld metallurgy,
6 and physical metallurgy.

7 So, Dr. Young, take it away.

8 DR. YOUNG: Thanks, John.

9 So, in Chapter 2 of the Topical Report, we
10 talk about why we chose 316H and the weld filler metal
11 16-8-2. These materials are currently qualified in
12 the ASME Code for high-temperature applications, and
13 as material is qualified, then components can be
14 designed, as John mentioned, for extremely low
15 probability of abnormal leakage and resistance to
16 rapidly propagating failure, as well as resistance to
17 gross rupture.

18 We know from its tremendous industrial
19 use, including use in nuclear power systems, 316H and
20 its weld metals exhibit desirable mechanical
21 properties, and they have some -- and growing, I'll
22 say -- demonstrated capability and compatibility with
23 Flibe salt and an extensive experience base.

24 Also, I'll just note that 316H and its
25 weld materials are used in other critical industries,

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1 like oil and gas and the chemical process industry, in
2 critical applications at the times and temperatures
3 relevant that we're designing components with for the
4 KP-FHR.

5 CHAIR PETTI: George?

6 DR. YOUNG: Yes?

7 CHAIR PETTI: This is Dave Petti. I just
8 have a question. I'm trying to understand.

9 I've sat through more presentations than
10 I can remember where experts from Oak Ridge said that
11 316 corroded too much in salt, and that was why Oak
12 Ridge developed Astralloy X, you know, after the MSRE
13 experience.

14 Now, I understand that Astralloy X isn't
15 Code-qualified, but where is the disconnect? You
16 guys, obviously, think 316H is acceptable. Is it an
17 issue of just lifetime, how to justify the two views?

18 DR. YOUNG: You know, I've thought about
19 this, also, Dave. And Oak Ridge published some
20 corrosion data for 316, and it wasn't too bad. And I
21 always wondered if maybe just they had the resources
22 to do long-term alloy development and wanted to do
23 that. But we see no limitation corrosion, or
24 otherwise, with using 316H. And we'll talk about some
25 of those data.

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1 CHAIR PETTI: Yes. No, it's a lot cheaper
2 than Astralloy.

3 DR. YOUNG: Yes.

4 CHAIR PETTI: But, yes, okay. I mean, I'm
5 glad you came to the same conclusion because that
6 disconnect has always been in the back of my mind.
7 So, thanks.

8 DR. YOUNG: Yes. I mean, I don't know why
9 Oak Ridge did it. If I could do alloy development,
10 that's fun work, but it's not really needed.

11 CHAIR PETTI: Yes, yes. Okay. Thanks.

12 DR. YOUNG: And then, in Chapter 3.1, we
13 talk about testing required for the ASME Code
14 extension, and we note that the base metal 316H is
15 qualified up to 816 C, but the weld filler metal that
16 we've chosen to match the properties there, 16A2,
17 that's currently only qualified to 650 C. So, Kairos
18 Power has taken on this Code extension, and we will
19 fully test that -- tensile test, creep fatigue, and
20 creep rupture testing -- to get that to match the base
21 metal up to 816 C.

22 3.2 talks about testing now to facilitate
23 both the Hermes and the commercial power reactor
24 designs. These are tests being conducted for the
25 stainless steels, for the ASME high-temperature design

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1 code model calibration and validation. And these are
2 all the tests we're conducting in high-temperature air
3 to do that. And those include tensile testing, stress
4 relaxation testing, exchange rate change or sometimes
5 called stress dip tests, uniaxial creep testing, as
6 well as creep fatigue testing.

7 And then, 3.3, some more high-temperature
8 air testing. And just a couple of notes. Based on
9 good feedback from the NRC, we've added a testing
10 program to look at stress through oxidation cracking
11 of 316H heat-affected zones relative to an alloy
12 that's known to be susceptible, specifically, alloy
13 347.

14 And then, as we'll talk about in a little
15 bit more detail, things like thermal stresses and
16 thermal striping were brought up in the metallics PIRT
17 as a potential concern. Those are currently managed
18 by design methods, and we feel that there's no further
19 testing needed to take those potential issues off the
20 plate.

21 Next slide.

22 And then, maybe really the meat of the
23 report is on environmental compatibility, and we talk
24 about the use of the PIRT data and the PIRT process.
25 I think we had a fantastic PIRT and it took a somewhat

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1 conservative view toward what we would assess. And
2 we'll talk about that.

3 The PIRT phenomena are grouped into four
4 major categories: corrosion, environmentally-assisted
5 cracking, kind of metallurgical effects or "other"
6 section, and finally, irradiation effects. And I'll
7 talk about this in a little bit more detail as we go
8 to the tables in a couple of slides.

9 Just to highlight, we've had a lot of very
10 good, very technical exchange with the NRC staff.
11 Some of the major comments that have been addressed
12 are as follows:

13 -----

14 We've added a lot of heat-affected zone
15 samples throughout the EAC testing and corrosion
16 testing. And these samples are kind of prioritized
17 for the non-power Hermes reactor.

18 As I mentioned, we added a pretty
19 significant direct testing program for the concern
20 over stress relaxation cracking, and we're going to do
21 a comparison study between 347 heat-affected zones,
22 where it's of known concern, and a 316H sample set,

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1 where it's a potential concern.

2 And then, we've also added a lot of detail
3 to the testing plans -- numbers of samples,
4 replicates, how we treat the data, exposure times, et
5 cetera.

6 CHAIR PETTI: George?

7 DR. YOUNG: Yes?

8 CHAIR PETTI: This is Dave again.

9 Just another question on the scope. I
10 understand this is good for the vessel and the piping.
11 But, like the control logs, do they have some sort of
12 metallic enclosure that they sit in? Is that sort of
13 outside the scope of this, or how does that --

14 DR. YOUNG: Yes, so we focused on the
15 vessel as the main safety-related component to contain
16 fission products.

17 And, John, do you want to address our
18 thoughts on how we're going to address concerns with
19 the control rods? We think we're going to treat that
20 separately.

21 MR. PRICE: Yes, it will be treated
22 separately.

23 CHAIR PETTI: Okay. No, that's good.
24 Just so I know where the scrim line is.

25 MEMBER KIRCHNER: Dave, that was my

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1 earlier question.

2 CHAIR PETTI: That's what I thought.

3 MEMBER KIRCHNER: Yes.

4 CHAIR PETTI: I just was going to follow
5 up just to make it clear at least in my mind.

6 MEMBER KIRCHNER: Yes.

7 CHAIR PETTI: Because there you could get
8 more than .1 BPA on the materials and it becomes more
9 complex then.

10 MEMBER KIRCHNER: Right. Yes.

11 CHAIR PETTI: Okay. Thanks.

12 MEMBER BALLINGER: This is Ron Ballinger.

13 Maybe this is for the closed session, but
14 I understand, and I see data on 321 and 347 on
15 cracking, but I haven't seen any data on 316H on that
16 type of cracking. Have you seen it?

17 DR. YOUNG: The AGR, the gas reactor
18 experience, there is some stress relaxation cracking
19 in 316H heat-affected zones. As best as I can tell,
20 it's in fairly atypical heats, high-sulfur heats,
21 somewhat abused material where you get significant
22 strain hardening from welding, probably some
23 segregation. And then, all those factors stack up.
24 And the British have seen stress relaxation cracking
25 in some 316H.

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1 MEMBER BALLINGER: Okay. Thanks.

2 DR. YOUNG: All right. So, just to
3 summarize, we've selected 316H base metal and ER16-8-2
4 weld filler metal as the metallic structural alloys
5 for use in our safety-significant, high-temperature,
6 Flibe-wetted components.

7 We're doing testing to support licensing
8 of both our non-power test reactor Hermes -- that's
9 got a four-year, full-power life, so kind of four-
10 plus-one hot commissioning -- and the commercial power
11 reactor with a xxxxxxxxxxxxxxxx KPX.

12 We're focusing the testing on these two
13 alloys for the reactor vessel. And again, that's our
14 primary safety-related component.

15 As you've noted from some of these
16 schematics, most all the penetrations are up in the
17 head. We don't have any low penetrations in the
18 vessel.

19 We're going to both extend the Code
20 qualifications for weld metal 16-8-2 to match the base
21 metal, and then, the materials testing effort to look
22 at potential environmental effects involves high-
23 temperature air testing for ASME design, and then,
24 testing in Flibe salt to account for any corrosion,
25 stress corrosion, et cetera.

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1 Any further questions on that?

2 CHAIR PETTI: Yes.

3 DR. YOUNG: Yes?

4 CHAIR PETTI: Just one more question. The
5 Topical talks about that you may clad certain areas
6 for enhanced corrosion resistance. Is that just sort
7 of risk-reduction activity that Kairos is pursuing,
8 you know, in the event that the testing doesn't go as
9 you anticipate, based on all the data that you've got
10 in the Topical thus far?

11 DR. YOUNG: Exactly. We thought for years
12 ago that we should leave open that possibility. We
13 had a little GAIN project that looked at cladding.
14 But, as we'll touch on in the closed session, we don't
15 have any plans or any need to --

16 CHAIR PETTI: Okay. I mean, I have sort
17 of a Code question. Were you to clad stuff, does that
18 remove the need for environmental testing because your
19 structure doesn't see the coolant?

20 DR. YOUNG: I don't think so. I think
21 those clad rules are drafted and being developed. And
22 so, there's a compliant clad and hard clad, and
23 certainly some benefit you can take there. I don't
24 think you could get away with no --

25 CHAIR PETTI: No testing? Yes.

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1 MEMBER BALLINGER: But I can answer that.
2 No, you can't.

3 MEMBER KIRCHNER: You can, yes.

4 MEMBER BALLINGER: No.

5 MEMBER KIRCHNER: The BWRs --

6 MEMBER BALLINGER: You have to assume that
7 the cladding has a breach.

8 CHAIR PETTI: Ah, okay.

9 MEMBER BALLINGER: So, you can't get away
10 with that.

11 CHAIR PETTI: Yes. Okay. Thanks.

12 MEMBER BALLINGER: And, you know, from my
13 perspective, the fact that you're going to qualify
14 these materials in all respects to Division 5, to
15 Section III, Division 5, and everything proceeds from
16 that, right?

17 DR. YOUNG: Right.

18 Any other questions or thoughts?

19 MEMBER KIRCHNER: George?

20 DR. YOUNG: Yes.

21 MEMBER KIRCHNER: This is Walt Kirchner
22 again.

23 Just could you address in the closed
24 session fasteners and other internals?

25 DR. YOUNG: Sure.

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1 MEMBER BALLINGER: Okay. Is that the
2 extent of the Kairos presentation?

3 DR. YOUNG: Yes, it is.

4 MEMBER BALLINGER: In the non-closed
5 session? Excuse me.

6 DR. YOUNG: Yes.

7 MR. PRICE: Yes.

8 MEMBER BALLINGER: Okay. Are there any
9 questions for Kairos from the members or consultants
10 before we switch over to the staff?

11 (No response.)

12 Hearing none, okay. We are proceeding at
13 a rapid rate. Can we get the staff presentation up?

14 MR. RIVERA: Yes. This is Richard Rivera,
15 Project Manager for this Topical Report. Give me one
16 moment to switch to the presentation.

17 So, can someone confirm if you can see the
18 slides?

19 MEMBER BALLINGER: Very well. Thank you.

20 MR. RIVERA: Excellent.

21 So, I will pass on the mic to John
22 Honcharik, who is the lead reviewer for this Topical
23 Report.

24 MR. HONCHARIK: All right. Good
25 afternoon.

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1 I'm John Honcharik, Senior Materials
2 Engineer in the Division of New and Renewed Licensing.
3 Alex Chereskin is in the Division of Advanced New
4 Reactors. And both of us reviewed the Topical Report
5 for Metallics Qualification and we'll present to you
6 our evaluation and conclusion of that Topical Report.

7 Next slide.

8 So, the NRC staff has reviewed the Topical
9 Report which provides the qualification plan for
10 metallic structural materials used in Flibe-wetted
11 areas for the safety-related, high-temperature
12 components of the KP-FHR power and test reactors.

13 The planned material testing, including
14 analysis and monitoring programs, will be used to
15 address the material reliability and compatibility of
16 the metallic materials in an environment of these
17 designs.

18 The results of these planned testing and
19 analysis will be used in future licensing applications
20 and provide a detailed description of the design,
21 inspection, and surveillance programs for the KP-FHR
22 design in order to demonstrate the materials'
23 reliability.

24 Next slide.

25 So, these are the regulations, and they

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1 describe what information would be in an application,
2 such as a construction permit, which would include the
3 materials and its compatibility to the design
4 environment, as described in this Topical Report.

5 Also, the Topical Report KP-TR-003
6 provides the NRC-approved principal design criteria
7 for the KP-FHR design. The PDCs below are applicable
8 to the qualification of the metallic components for
9 the KP-FHR design which the staff based this review
10 on. The two are PDC 14 and 31, which, basically,
11 relate to ensure that the reactor coolant pressure
12 boundary have an extremely low probability of abnormal
13 leakage, rapidly propagating failure, and to fail in
14 a non-brittle manner, and to minimize the probability
15 of rapidly propagating failure.

16 Next slide.

17 The staff's review focused on the overall
18 testing framework, to include if there is reasonable
19 assurance of the testing for environmental effects of
20 Flibe on the metallic structural materials provided in
21 Section 4 of this Topical Report, and that will also
22 be discussed today, which are: the material, which
23 are 316H and its fill metal ER-16-8-2; the test
24 environment as it relates to these designs, and the
25 four degradation mechanism categories of corrosion,

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1 environmentally-assisted cracking, effects of
2 metallurgical properties, and irradiation.

3 Next slide.

4 Within the four degradation categories
5 noted previously, 14 types of degradation mechanisms
6 were proposed to be evaluated by testing, analysis, or
7 surveillance. How these degradation mechanisms were
8 addressed in the qualification program will be
9 discussed later in the presentation.

10 Next slide.

11 First, we'll discuss the material to be
12 used. The metallic structural materials proposed for
13 the KP-FHR designs are 316H, also known as stainless
14 steel, and the associated stainless steel weld fill
15 metal ER-16-8-2. These materials are qualified for
16 use in ASME Code, Section III, Division 5, for high-
17 temperature reactors with respect to their mechanical
18 properties. Division 5 of Section III will be
19 endorsed in Reg Guide 1.87, Revision 2, and will be
20 published soon.

21 There is Condition 4 related to the weld
22 material mechanical properties, as George alluded to
23 earlier, since the fill metal is not currently
24 qualified to the higher temperatures necessary to
25 support the accident scenarios of these designs. So,

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1 therefore, the NRC imposed the condition that the fill
2 metal must be qualified to that temperature, in
3 accordance with the requirements of Division 5 that
4 balance the postulated accident conditions, and
5 approved by the NRC.

6 The staff's review of the Topical Report,
7 as presented today, will only address the
8 environmental effects of Flibe on the metallic
9 material, since the mechanical properties are already
10 covered in Division 5 for the ASME Code.

11 Next slide.

12 Next, we will discuss the test environment
13 used for the proposed material testing. The staff
14 found that the material testing environment duplicates
15 the operating environment of the KP-FHR designs listed
16 above. Higher test temperatures are used in some
17 tests in order to develop environmental degradation
18 rates during postulated accident scenarios.

19 Testing conducted in nominal Flibe which
20 is applied with minimal water and contaminants, the
21 testing in this nominal Flibe will provide more
22 conservative results since the nominal Flibe is more
23 oxidizing and will produce greater corrosion rates.
24 Additional testing will be performed in redox
25 controlled Flibe to see if these corrosion rates can

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1 be minimized.

2 There is Condition and Limitation No. 3
3 related to the environment to be tested. The staff
4 imposed a condition that, if the time and temperature
5 for both normal operation and postulated accident
6 conditions change for these designs, they must still
7 be bounded by the NRC staff-endorsed ranges found in
8 table 2 of Reg Guide 1.87.

9 Next slide.

10 Concerning the postulated accident
11 scenarios, there are two potential accident scenarios
12 for the commercial power reactor. Those are
13 intermediate salt ingress and, also, air ingress into
14 the Flibe. For the non-power test reactor, there is
15 one potential accident scenario, which is an air
16 ingress into the Flibe.

17 These impurity ingresses would produce a
18 specific concentration of these impurities that could
19 affect a safety-related component. So, the NRC finds
20 this approach of testing with these impurities that
21 simulate these accident conditions acceptable.

22 For the power reactor, testing with both
23 air and intermediate salt as impurities is reasonable
24 to develop the effects of impurities on the corrosion
25 rate on 316H and its fill metal because it will bound

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1 the accident conditions for the power reactor.
2 Likewise, for the test reactor, it will be done with
3 air as an impurity, which we found acceptable also.
4 As part of this testing, Condition 3, which I stated
5 previously in a previous slide, would be applicable.

6 In addition, Condition 8 would apply,
7 since the details of this impurity testing, that is,
8 the concentration of each of the contaminants, have
9 not been determined. Therefore, the specific
10 concentration of each contaminant used in impurity
11 effects testing on 316H and its fill metal shall bound
12 the accident scenarios postulated in the transient
13 analysis for these KP-FHR designs.

14 The next topic reviewed for the Topical
15 Report will be the degradation mechanisms. The
16 following slides will discuss each degradation
17 mechanism accounted for in the qualification program.

18 The first is corrosion. This includes the
19 various types of corrosion, such as general corrosion,
20 crevice corrosion, thermal aging, and erosion and
21 wear. The Topical Report proposes that the corrosion
22 testing will be used to develop quantitative corrosion
23 models for 316H and its fill metal in the Flibe
24 environment.

25 The proposed testing will use

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1 representative coupons of these materials and
2 conditions described in tables 12 and 13 of the
3 Topical Report, and will be performed under different
4 conditions to assess the impact of these specific
5 corrosion degradation mechanisms.

6 Next slide.

7 So, the NRC found the proposed testing
8 acceptable to determine the impact Flibe has on the
9 corrosion rates of the material, based on the
10 following: the impact of temperature; microstructure;
11 the salt composition, which would include the nominal
12 redox and impurity ingresses; the geometry;
13 erosion/corrosion, thermal aging, and presence of
14 graphite and redox control on the corrosion rate of
15 316H and its fill metal, which can be determined from
16 these proposed tests.

17 For example, the NRC finds the proposed
18 tests acceptable for simulating the occlusal or
19 crevice geometry effects in order to determine whether
20 or not crevice corrosion is a concern for these
21 materials in Flibe.

22 Tests for the effect of temperature on the
23 corrosion rate would also be acceptable because the
24 corrosion is evaluated over the range of temperatures,
25 consistent with both operating temperatures and

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1 postulated bounding accident conditions.

2 Next slide.

3 Continued acceptability of the impacts of
4 corrosion. Tests are consistent with the expected
5 corrosion mechanisms for 316H in a molten salt
6 environment. In addition, some tests will be
7 conducted with foam Flibe, which is necessary in order
8 to simulate the flowing salt in the reactor.

9 Graphite particles will be used in tests
10 with flow in order to determine the effects of these
11 particles on rates of erosion/corrosion.

12 In addition, the effects caused by
13 microstructure changes due to weld, heat-affected
14 zones, thermal aging, and cold working on the
15 corrosion rates of the material will also be tested
16 and evaluated.

17 Next slide.

18 The next degradation category is for
19 environmentally-assisted cracking. This includes
20 stress corrosion cracking, corrosion fatigue, and
21 environmental creep.

22 For stress corrosion cracking and
23 corrosion fatigue, the NRC finds the proposed test
24 plan provides reasonable assurance in determining the
25 crack growth rates for these two mechanisms in the

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1 environment of the KP-FHR designs, because in situ
2 mechanical testing will be performed using slow strain
3 rate testing for corrosion fatigue and stress
4 corrosion cracking in both normal and accident
5 conditions. The SSRT testing will be conducted in
6 nominal Flibe and redox controlled Flibe to assess if
7 316H and its fill metal are susceptible to stress
8 corrosion cracking. The samples to be tested are
9 representative of the material and weldments to be
10 used in the KP-FHR designs, and therefore, are
11 acceptable. The testing methodology will also be
12 conducted in accordance with well-established ASTM
13 specifications such as G129.

14 Next.

15 For environmental creep degradation, the
16 NRC found the proposed testing plan acceptable because
17 pretesting in both nominal Flibe and in air using
18 welded base metal samples, to include both base metal,
19 weld metal, and heat-affected zones, ensures each
20 material's state is tested in the operating
21 environment.

22 Creep tests will be performed in both air
23 and in nominal Flibe. Tests in nominal Flibe will
24 determine if the Flibe contributes any additional
25 degradation beyond those determined from the creep

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1 tests performed in air. Additional testing will be
2 required to quantify any increase in degradation
3 contributing to the Flibe if the testing determines
4 Flibe has this additional effect.

5 Next slide.

6 The next degradation category is for
7 metallurgical effects, as designated in the Topical
8 Report. This includes the degradation mechanism of
9 stress relaxation cracking, phase formation
10 embrittlement, and thermal cycling.

11 For stress relaxation cracking, the NRC
12 found the proposed testing plan acceptable. Both the
13 current data and experience shows that only the heat-
14 affected zone of the 316H base metal with high
15 triaxial stresses are susceptible. Testing in air is
16 acceptable since the triaxial stresses are the major
17 contributors to the stress relaxation cracking and not
18 the environment.

19 Comparing the susceptibility of 316H to
20 that of 347, which is known to be susceptible, would
21 allow a determination of the bounding triaxial
22 stresses.

23 Test results will be used, then, to
24 conduct further analysis and design requirements of
25 the KP-FHR design, such as weld designs and specific

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1 weld processes and parameters, in order to reduce
2 these triaxial stresses, and thereby, minimizing
3 stress relaxation cracking.

4 Next slide.

5 For phase formation embrittlement, the NRC
6 found the proposed testing plan acceptable because
7 phase formation embrittlement may occur when 316H and
8 its fill metal picks up an element during its exposure
9 to Flibe and forms a deleterious second phase.
10 Testing will determine whether this degradation
11 mechanism occurs for the KP-FHR design environment.

12 As part of this testing, Condition 11 is
13 applicable, which states that, if intermetallic
14 formation occurs, an applicant will need to perform
15 testing to quantify the effects on the mechanical
16 properties of 316H and its fill metal.

17 For thermal cycling --

18 MEMBER KIRCHNER: Richard, this is Walt.

19 Just historically, what elements would
20 contribute to that last phase embrittlement
21 phenomenon? When you say, "elements," are talking
22 trace fission products, erosion or corrosion products
23 from the metals?

24 MR. HONCHARIK: Yes, this is John
25 Honcharik.

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1 I think right now, for the open session,
2 I'm not sure I could say any more. We'd probably have
3 to answer that in the closed session. So, I can make
4 a note of that.

5 MEMBER KIRCHNER: That's fine. Thank you.

6 MR. HONCHARIK: Yes.

7 Okay. So for thermal cycling, the NRC
8 staff found the proposed testing plan acceptable
9 because large thermal transients could lead to high
10 stresses, resulting in thermal fatigue degradation of
11 316H and its fill metal. These thermal stresses will
12 be addressed by conducting analysis in order to refine
13 the design and operation of the KP-FHR designs to
14 mitigate these large thermal gradients.

15 As part of this analysis, Condition and
16 Limitation 12 apply and the applicant will assess the
17 thermal cycling and striping in future licensing
18 submittals by minimizing the thermal gradients via
19 appropriate design and operating conditions.

20 Next slide.

21 The next degradation category is for
22 irradiation-induced embrittlement. This includes
23 irradiation-induced embrittlement, irradiation-
24 affected corrosion, and irradiation-assisted stress
25 corrosion cracking.

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1 For irradiation-induced embrittlement, the
2 NRC staff found the proposed testing plan acceptable
3 because the data shows that irradiation-induced
4 embrittlement at low strain rates can affect the
5 material properties, such as tensile strength, (audio
6 interference), and creep (audio interference) due to
7 generation healing.

8 Degradation factors will be developed
9 based on these existing data. However, irradiation
10 tests will be conducted on 316H and its fill metal,
11 including the heat-affected zone of the 316H, in order
12 to quantify any design margins at the irradiation
13 levels posed for both the test reactor and the
14 commercial reactor. The details of that testing will
15 be provided in a future application.

16 As part of this testing, Limitation and
17 Condition 13 would apply and that the test environment
18 shall bound the KP-FHR designs, including expected
19 irradiation damage and healing content.

20 For irradiation-affected corrosion, the
21 NRC staff found the proposed testing plan acceptable
22 because existing data shows that irradiation may
23 increase general corrosion rates. However, the
24 materials surveillance program for the non-power test
25 reactor and commercial power reactor will be used to

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1 monitor irradiation-affected corrosion. In addition,
2 an inspection and monitoring program will assess the
3 wall thickness of the reactor vessel and will be
4 implemented. So, therefore, existing data will be
5 used in order to develop the degradation factors and
6 be monitored by the surveillance program.

7 And as part of that surveillance program,
8 Condition 14 would apply, and that this program,
9 including the inspection and monitoring program, must
10 be implemented for the test reactor and commercial
11 power reactor. You will use the KP-FHR designs to
12 assess and monitor both irradiation-affected corrosion
13 rates and, also, for stress corrosion cracking, which
14 we'll talk about in the next slide.

15 So, for irradiation-assisted stress
16 corrosion cracking, NRC found the proposed testing
17 plan acceptable because irradiation-induced stress
18 corrosion cracking is not expected at the low
19 irradiation level. The stress corrosion cracking test
20 program specified in Section 4.24 will determine if
21 stress corrosion cracking is a credible degradation
22 mechanism for the environment in these designs.

23 Therefore, the material surveillance
24 program for both the non-power test reactor and the
25 commercial power reactor will be used in order to

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1 monitor this degradation mechanism. And the condition
2 that I stated previously would also apply to this
3 surveillance program.

4 So, in conclusion, the NRC staff finds
5 that there's reasonable assurance that the material
6 testing plan, including the analysis, surveillance,
7 and monitoring, for both 316H and its associated fill
8 metal, with the limitations and conditions noted in
9 the Safety Evaluation, can be used to provide the
10 necessary information to address the materials
11 reliability and compatibility in the environment of
12 the KP-FHR designs.

13 And that's based on, in summary:

14 The testing duplicates the operating
15 environment for both normal and accident conditions.

16 The material test samples are
17 representative of actual weldments.

18 Analysis will be performed to mitigate
19 stress relaxation cracking and thermal cycling through
20 design and operations.

21 The material surveillance program will be
22 used to monitor for irradiation effects on corrosion
23 and stress corrosion cracking.

24 And the results of the planned tests and
25 analysis will be used for future licensing

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1 applications to ensure a component's performance
2 safety function and that there would be an extremely
3 low probability of abnormal leakage propagating
4 failure or brittle failure, which would partially
5 PDCs 14 and 31.

6 And I have it open for questions.

7 MEMBER BALLINGER: Questions from the
8 members and/or consultant?

9 (No response.)

10 Hearing none, before we switch to the
11 closed session, we'll open the line and ask for public
12 comment.

13 If there are members of the public that
14 would wish to make a comment, please unmute yourself
15 or press *6, and make your comment.

16 (No response.)

17 Hearing none, thank you very much for your
18 presentation.

19 We've been going very fast. So, I would
20 suggest that we take a break until 3:00 p.m., and
21 then, come back on in the closed session. So, we'll
22 take what amounts to about a 10-minute break, and
23 then, we'll see you in the closed session at 3:00 p.m.

24 (Whereupon, the above-entitled matter went
25 off the record at 2:52 p.m.)

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KP-NRC-2301-001

Enclosure 2
Presentation Materials for the January 12, 2023, ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)




Kairos Power

Metallic Materials Qualification Topical Report

ACRS Kairos Power Subcommittee Meeting

January 12, 2023

OPEN SESSION



Kairos Power's mission is to enable the world's transition to clean energy, with the ultimate goal of dramatically improving people's quality of life while protecting the environment.

Introduction

- This report presents the methods for qualifying metallic materials for use in KP-FHRs
 - Qualification is subject to the conditions in topical report
 - Uses KP-X PIRT test plan as a baseline but with reduced test durations for the shorter-lived Non-Power Test Reactor
- Demonstration of qualification will be documented in safety analysis report documents as part of licensing applications under 10 CFR Part 50 or Part 52
- This report is applicable to the Non-Power Test Reactor (Hermes) and to the Commercial Power (KPX) Reactor



Summary of Key Parameters in KP-FHR (Table 1)

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	

Metallic Materials Topical Report Organization

- Introduction
 - KP-FHR Technologies
 - Regulatory Information
- Structural Alloys
 - Background
 - Structural Alloy Selection
 - Industrial Experience
- Air Testing and Finite Element Analysis
 - Testing Required for ASME Code Extension
 - Testing to Facilitate Reactor Designs
 - High Temperature Testing & Analysis to Support Potential Degradation
- Compatibility with Flibe and Irradiation
- Environmental Compatibility
- Conclusions and Limitations
- Appendix A: Coatings, Cladding, and Tritium Management
- Appendix B: Inspection and Aging Management
- Appendix C: Details of the Corrosion Data Analysis
- Appendix D and E: Not Used
- Appendix F Certified Material Reports

Chapter 2: Alloy 316H and Weld Filler Metal ER16-8-2

- Alloy 316H and Weld Filler Metal ER16-8-2 were chosen because of its qualification in high temperature applications. As the material is qualified, components can be designed for extremely low probability of abnormal leakage, resistance to rapidly propagating failure, and resistance to gross rupture.
- Alloy 316H and its weld metals exhibit desirable mechanical properties, have demonstrated compatibility with Flibe salt, and have an extensive experience base in nuclear applications.
- Alloy 316H and its weld metals are used in other industry applications near the time and temperature of the KP-FHR.

Chapter 3.1: Testing Required for ASME Code Extension

- Alloy 316H is qualified for 816°C and Weld Filler Metal is currently qualified to 650°C
- The weld filler metal will be tested to extend the ASME qualification of ER16-8-2 to match the base metal.
 - Elevated Temperature Tensile Testing
 - Creep-Fatigue Testing
 - Creep-Rupture Testing

Chapter 3.2: Testing to Facilitate Non-Power and Commercial Power Reactor Designs

- The types of tests being conducted for Alloy 316H stainless steel model calibration and validation of ASME design methodologies (all conducted in high temperature air)
 - Tensile Testing
 - Stress Relaxation Testing
 - Strain Rate Change (Stress Dip) Testing
 - Uniaxial Creep Testing
 - Creep-Fatigue Testing

Chapter 3.3: High Temperature Testing & Analysis to Support Potential Degradation

- Stress Relaxation Cracking – testing program per discussion with NRC Staff
- Thermal Stresses & Thermal Striping – managed via design

Chapter 4.2: Environmental Compatibility

- Use of the PIRT Data
- Corrosion
 - Corrosion Testing with Use of Compositional Analysis and Electrochemical Potential (ECP) Monitoring
- Environmentally Assisted Cracking
 - Slow Strain Rate Testing (SSRT)
 - Fracture Mechanics Based Testing – Corrosion Fatigue (CF) and Stress Corrosion Cracking (SCC)
 - Environmental Creep Testing
- Metallurgical Effects / Other
 - Stress Relaxation Cracking
 - Phase Formation Embrittlement
 - Thermal Cycling / Striping
- Irradiation Effects
 - Irradiation-Induced Embrittlement
 - Irradiation-Affected Corrosion
 - Irradiation-Assisted Stress Corrosion Cracking (IASCC)

Some Major Comments that Have Been Addressed

- Testing at 750°C added to bound maximum expected transient temperature and time
- Heat affected zone (HAZ) samples added throughout EAC and Corrosion testing and these samples are priorities for Hermes Rx testing
- Direct testing for stress relaxation cracking added – comparison study between 347 HAZ and 316H HAZ
- Significant detail added to testing plans (e.g., numbers of samples, exposure times, etc. for corrosion tests, specific conditions for environmentally assisted cracking tests)

Summary

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

NRC Evaluation of KP-TR-013-P, “Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)”, Rev. 4

John Honcharik
Alex Chereskin

US Nuclear Regulatory Commission

January 12, 2023

Introduction

- Kairos Power, LLC requested staff review and approval of Topical Report (TR) KP-TR-013-P, Rev. 4, “Metallic Material Qualification for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)” to address environmental effects on materials
- TR provides a material testing plan (including analysis and monitoring programs) 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
- These material qualification test results will be used as a basis in future licensing actions to address potential materials reliability and environmental compatibility in KP-FHR designs¹ via design, operation, inspection and surveillance programs to manage the materials performance.

¹ The term “KP-FHR designs” applies to both the power reactor and non-power test reactor, unless otherwise specified.

Regulatory Basis

Title 10 of the *Code of Federal Regulations* (10 CFR) Sections 50.34(a), 50.34(b), and corresponding regulations for design certification applications, combined license applications and standard design approvals

The following Kairos Principal Design Criteria (PDC) are applicable to this topical report and were previously approved by the NRC staff (KP-TR-003-NP-A):

KP-FHR PDC 14, “Reactor coolant boundary”

KP-FHR PDC 31, “Fracture prevention of reactor coolant boundary”

Staff Evaluation-Overview

- The staff's review focused on the overall testing framework to conclude that there is reasonable assurance that the test plan, in part, will satisfy KP-FHR PDCs 14 and 31. Specific topics included:
 - Material - 316H and ER-16-8-2 stainless steel
 - Test environment
 - Four degradation mechanism categories:
 - Corrosion
 - Environmentally assisted cracking
 - Effects on Metallurgical properties
 - Irradiation

Staff Evaluation-Overview

14 Degradation mechanism addressed:

- Corrosion
 1. general corrosion
 2. crevice corrosion
 3. thermal aging
 4. erosion/wear
 5. cold leg occlusion
- Environmentally assisted cracking
 6. stress corrosion cracking
 7. environmental creep
 8. corrosion fatigue
- Effects on Metallurgical properties
 9. stress relaxation cracking
 10. phase formation embrittlement
 11. thermal cycling
- Irradiation
 12. irradiation-affected corrosion,
 13. irradiation-assisted stress corrosion cracking,
 14. irradiation-induced embrittlement

Staff Evaluation-Material

- Metallic Material Used

- 316H austenitic stainless steel and associated weld material ER-16-8-2 stainless steel
- Qualification for mechanical properties of 316H and ER16-8-2 weld material are provided by ASME Code, Section III, Division 5, and endorsed in Regulatory Guide 1.87, Rev. 2 (TBA))
- Both materials are listed in Section III, Division 5 for high temperature applications (Limitation & Condition 4 - ER16-8-2 must be qualified to postulated accident conditions)
- TR only addresses environmental effects and does not cover the mechanical properties qualification

Staff Evaluation-Environment

- KP-FHR Environment (Operating)
 - Flibe salt temperatures of 550°C-650°C
 - An intermediate salt coolant loop (commercial reactor)
 - A Primary Heat Transport System to the air (non-power test reactor)
 - Non-power test reactor lifetime of 5 years and commercial power reactor lifetime
 - “Near-atmospheric” primary coolant pressures
 - End of life irradiation of less than 0.1 displacement per atoms (dpa)
- Testing environment duplicates the environment of KP-FHR designs.
- Higher test temperature to develop environmental degradation rates during postulated accident scenarios
- Testing conducted in “Nominal Flibe”, (Flibe purified to minimize water and other oxidizing contaminants but not metal additions to invoke redox control)
- Additional testing in Redox Controlled Flibe
- Limitation & Condition 3 – Time and temperature must bound RG 1.87, Table 2

Staff Evaluation-Environment

KP-FHR Environment (Postulated accident scenarios)

- Two potential accident scenarios for the commercial power reactor (i.e., intermediate salt ingress and air ingress for into the Flibe salt)
- Potential accident scenario for non-power test reactor (Nominal Flibe with air as an impurity)
- The NRC staff finds testing with impurities acceptable
 - Power reactor – testing of 316H and ER16-8-2 in Nominal Flibe with air and intermediate salt (as an impurity) will bound accident conditions and provides a reasonable method of developing corrosion rates.
 - Non-power reactor - corrosion testing of 316H and ER16-8-2 in Nominal Flibe with air (as an impurity) provides a reasonable method of developing corrosion rates for the non-power test reactor.
- Limitation & Conditions 3 - Time and temperature must bound RG 1.87, Table 2
- Limitation & Condition 8 - Impurity ingress testing (e.g., the concentration of contaminant) must bound safety analysis

Staff Evaluation-Corrosion

Corrosion

- Proposed corrosion testing will be used to develop quantitative corrosion models for 316H and ER16-8-2 stainless steel in a Flibe environment.
- Proposed testing will use representative coupons of these materials in conditions described in Tables 12 and 13 of the TR.
- Tests performed under different conditions to assess the impacts of specific corrosion degradation mechanisms.

Staff Evaluation-Corrosion

- The NRC staff found proposed qualification testing acceptable to determine the impacts of corrosion
 - These tests will determine the impact of temperature, microstructure, salt composition, geometry, erosion-corrosion, thermal aging, presence of graphite, and redox control on the corrosion rates of 316H and ER16-8-2.
 - Occluded geometry effects on corrosion rates -acceptable because tests will determine whether crevice corrosion is a concern for 316H and ER16-8-2 in Flibe
 - Effect of temperature on corrosion rates - acceptable since tests are evaluated over a range of temperatures consistent with the operating temperatures of the KP-FHR designs including bounding postulated accident conditions

Staff Evaluation-Corrosion

- The NRC staff found proposed qualification testing acceptable to determine the impacts of corrosion (cont'd)
 - Tests are consistent with the expected corrosion mechanisms for 316H and ER16-8-2 in a molten salt environment and a portion of the tests will be conducted with flowing Flibe, which is necessary to simulate the flowing salt in a reactor.
 - 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
 - Impact of microstructure effects (heat affected zone, thermal aging and cold work) will be tested and evaluated.

Staff Evaluation-Environmentally Assisted Cracking

Stress Corrosion Cracking and Corrosion Fatigue

The NRC staff found the proposed testing plan provides reasonable assurance in determining the crack growth rates for fatigue and stress corrosion cracking relative to the environment in the KP-FHR designs:

- In-situ mechanical testing systems will be used to conduct slow strain rate testing (SSRT) for corrosion fatigue and stress corrosion cracking in both normal and accident conditions
- The samples to be tested are acceptable because they are representative of the material and weldments to be used in the KP-FHR designs.

Staff Evaluation-Environmentally Assisted Cracking

Environmental Creep

The NRC staff found the proposed testing plan acceptable:

- Creep testing in both Normal Flibe and air using welded base metal samples to include the base metal, weld metal and heat affected zone
- Creep tests in Nominal Flibe to determine if the Flibe contributes additional degradation beyond those determined 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

Staff Evaluation-Metallurgical Effects

Stress Relaxation Cracking

The NRC staff found the proposed testing plan acceptable:

- 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
- Testing in air is valid for 316H in Flibe for the KP-FHR designs since triaxial stresses are the major contributor to stress relaxation cracking and not environment
- Comparing susceptibility of 316H to that of 347 would allow a determination of the bounding triaxial stresses
- Using test results to conduct future analysis and design refinements of the KP-FHR designs, such as weld designs and specific weld processes and parameters to reduce triaxial stresses thereby minimizing stress relaxation cracking

Staff Evaluation-Metallurgical Effects

Phase Formation Embrittlement

The NRC staff found the proposed testing plan acceptable:

- 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
- Testing will determine whether this degradation mechanism occurs for the KP-FHR designs
- Limitation & Condition 11 - If intermetallic formation occurs, an applicant will need to perform testing to quantify the effects on the mechanical properties of 316H and associated weld material ER16-8-2

Staff Evaluation-Metallurgical Effects

Thermal Cycling/Striping

The NRC staff found the proposed testing plan acceptable:

- Degradation of 316H and ER16-8-2 by large thermal transients could lead to high stresses resulting in thermal fatigue degradation
- Thermal cycling will be addressed by conducting analysis to refine the design and operation of the KP-FHR designs to mitigate large thermal gradients.
- Limitation & Condition 12 - Assess thermal cycling/striping in future licensing submittals

Staff Evaluation-Irradiation Effects

Irradiation-Induced Embrittlement

The NRC staff found the proposed testing plan acceptable:

- At low strain rates, data shows irradiation-induced embrittlement can affect tensile strength, ductility and creep life due to the generation of helium.
- Existing data will be used to develop degradation factors
- 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
- Limitation & Condition 13 - Test environment shall bound the KP-FHR designs, including the expected irradiation damage (dpa) and helium content

Staff Evaluation-Irradiation Effects

Irradiation-Affected Corrosion

The NRC staff found the proposed testing plan acceptable:

- Existing data shows that irradiation may increase general corrosion rates
- Materials surveillance system program for the non-power test reactor and the commercial power reactor systems to monitor irradiation-affected corrosion.
- Inspection and monitoring program that will assess the wall thickness of the reactor vessel will also be implemented
- Existing data will be used to develop degradation factors and monitored by surveillance
- Limitation & Condition 14 - A materials surveillance program and an inspection and monitoring program must be implemented

Staff Evaluation-Irradiation Effects

Irradiation-Assisted Stress Corrosion Cracking

The NRC staff found the proposed testing plan acceptable:

- IASCC is not expected to be a degradation mechanism in the KP-FHR design due to the low irradiation level (<0.1 dpa)
- Stress corrosion 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
- Materials surveillance system program for the non-power test reactor and commercial power reactor systems to monitor IASCC.
- Limitation & Condition 14 - A materials surveillance program and an inspection and monitoring program must be implemented

Conclusions

- With limitations and conditions noted, material testing plan (including analyses, surveillance and monitoring) for 316H and ER16-8-2 will provide necessary information to address the materials reliability and compatibility in the environment of the KP-FHR designs:
 - Testing duplicates the operating environment (normal and accident) that the material will experience in the KP-FHR designs
 - Material test samples are representative of actual weldments
 - Analyses for stress relaxation cracking and thermal cycling
 - Surveillance for irradiation effects (corrosion and stress corrosion cracking)
 - Results of planned tests and analyses will be used for future license applications to ensure extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture, which partially satisfies PDC 14 and PDC 31

Questions?

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LIMITATIONS AND CONDITIONS (Redacted)

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.

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LIMITATIONS AND CONDITIONS (Redacted; Cont.)

4. (Section 4.2.1) ER16-8-2 material must be qualified to a temperature of [[
--]] 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.

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LIMITATIONS AND CONDITIONS (Redacted; Cont.)

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 [[]]
- “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 [[]], the test temperature and time for the associated material testing in Section 4 of the TR must be increased to [[]].

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LIMITATIONS AND CONDITIONS (Redacted; Cont.)

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
]] 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.

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LIMITATIONS AND CONDITIONS (Redacted; Cont.)

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.

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