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1 UNITED STATES OF AMERICA

2 NUCLEAR REGULATORY COMMISSION

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4 632ND MEETING

5 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

6 (ACRS)

7 + + + + +

8 THURSDAY

9 MARCH 3, 2016

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11 ROCKVILLE, MARYLAND

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13 The Advisory Committee met at the Nuclear

14 Regulatory Commission, Two White Flint North, Room

15 T2B1, 11545 Rockville Pike, at 8:30 a.m., Dennis Bley,

16 Chairman, presiding.

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1 COMMITTEE MEMBERS:
2 DENNIS C. BLEY, Chairman
3 MICHAEL L. CORRADINI, Vice Chairman
4 PETER RICCARDELLA, Member-at-Large
5 RONALD G. BALLINGER, Member
6 CHARLES H. BROWN, JR. Member
7 DANA A. POWERS, Member
8 HAROLD B. RAY , Member
9 JOY REMPE, Member
10 GORDON R. SKILLMAN, Member
11 JOHN W. STETKAR, Member
12
13 DESIGNATED FEDERAL OFFICIAL:
14 QUYNH NGUYEN
15
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1 ALSO PRESENT:
2 AL ADAMS, NRR
3 PHIL AMWAY, Exelon
4 ROBERT BEALL, NRR
5 ANTHONY BOWERS, NRR
6 GREGORY BOWMAN, NRR
7 RANDY BUNT, NEI*
8 HOSSEIN ESMAILI, NRC
9 ED FULLER, Office of Research
10 JEFF GABOR, Jensen Hughes
11 MIRELA GAVRILAS, NRR
12 DUANE HARDESTY, NRR
13 WALTER KIRCHNER, Invited Expert
14 STEVEN KRAFT, NEI
15 REED LABARGE, PWROG/WEC
16 MARVIN LEWIS, Public Participant*
17 JOSE MARCH-LEUBA, Invited Expert
18 DAMIAN MIRZIO, PWROG
19 WILLIAM RECKLEY, NRR
20 JOSEPH SEBROWSKY, NRR
21 MATTHEW SUNSERI, Invited Expert
22 ANDREA D. VALENTIN, Executive Director, ACRS
23
24 *Present via telephone
25

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1 P R O C E E D I N G S

2 8:30 a.m.

3 CHAIRMAN BLEY: The meeting will now come

4 to order.

5 This is the first day of the 632nd meeting

6 of the Advisory Committee on Reactor Safeguards.

7 During today's meeting, the Committee will

8 consider the following, Non-power Production and

9 Utilization Facilities Proposed License Renewal

10 Rulemaking, Biennial Review and Evaluation of the NRC

11 Safety Research Program, Group 2 Fukushima Tier 2 and

12 3 Closure Plans and Preparation of ACRS Reports.

13 This meeting is being conducted in

14 accordance with the provisions of the Federal Advisory

15 Committee Act.

16 Mr. Quynh Nguyen is the Designated Federal

17 Official for the initial portion of this meeting.

18 We have no written comments or requests to

19 make oral statements from members of the public

20 regarding today's sessions.

21 There will be a phone bridge line. To

22 preclude interruption of the meeting, the phone will

23 be placed in the listen in mode during the

24 presentations and Committee discussions.

25 A transcript of portions of the meeting is

being kept and it is requested that the speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so that they can be readily heard.

I also want to make you aware that this meeting is being webcast with the ability to view our presentation slides on the web. Those of you who are on the bridge line may want to do that, can dial into the bridge line or connect through the NRC's public meeting website and click on the link.

It does work. It works very well and from every one who's reported to us, the sound is much better than on the bridge line, much less noisy and clearer.

If you have trouble with it, you may call our office.

The Committee, at this time, would like to welcome Ms. Andrea Valentin as the new Executive Director for ACRS.

Andrea has been with NRC since 1992 and reached the position of Senior Materials Engineer before progressing through the management ranks.

Andrea?

(APPLAUSE)

CHAIRMAN BLEY: Andrea will be replacing

1 Dr. Edwin Hackett and the Committee would like to
2 thank him for his long and valued service. Thank you,
3 Ed.
4 (APPLAUSE)
5 CHAIRMAN BLEY: Not that you're abandoning
6 us or anything like that. We hope you come back to
7 visit.
8 Furthermore, the Committee would like to
9 acknowledge several subject matter experts in
10 attendance today.
11 Please welcome Dr. Walt Kirchner, Dr. Jose
12 March-Leuba and Mr. Matt Sunseri. And, they're here
13 somewhere. Over here.
14 (APPLAUSE)
15 CHAIRMAN BLEY: At this time, I will turn
16 the meeting over to Dr. Dana Powers for the first
17 session.
18 MEMBER POWERS: Thank you, Mr. Chairman.
19 Members are, of course, aware that not all
20 reactors in this world are designed to produce
21 kilowatts. Some of them actually serve research and
22 production and a variety of other tasks.
23 We are looking today at a potential
24 revision to the way we license, and to some extent,
25 monitor and inspect these non-power production and

utilization facilities.

Mirela, are you going to give us an introduction?

MS. GAVRILAS: I will be glad to, yes, thank you.

MEMBER POWERS: You have to.

MS. GAVRILAS: I will.

MEMBER POWERS: You want to.

MS. GAVRILAS: Absolutely.

MEMBER POWERS: So, I'll turn to Mirela to give us an introduction on this subject and tell us what we're embarked on here.

MS. GAVRILAS: Thank you, Dr. Powers.

I'm Mirela Gavrilas. I'm the WD Director for Research and Test Reactors in NRR.

So, like Dr. Powers said, the proposed rule we'll discuss today was a rule that the Commission asked us to engage in in order to prevent a potential recurrence of the backlog for relicensing research reactors.

We've also taken this opportunity to fix some deficiencies in our regulations, they're minor but very important and pertinent to the mandate from the Commission.

On February 3rd, we had the benefit of

1 interacting with the RTR Subcommittee. And, following
2 those discussions actually shaped the presentation
3 that you will see today.
4 I mentioned then and it's worth mentioning
5 again in this forum that, as you all know, the
6 Agency's going through a rebaselining exercise. And,
7 one of the things that were proposed for the
8 efficiencies list was rulemakings of medium priority,
9 this being among them.
10 After deliberation, not long deliberation,
11 but deliberation, everybody agreed to keep this rule
12 going for two reasons.
13 One, it's quite advanced, but more
14 importantly, this rule is exactly in the spirit of
15 efficiency and effectiveness.
16 As you'll see in the presentation of the
17 regulatory analysis, it lifts the burden on the
18 licensee tremendously and it also lightens the burden
19 on the staff.
20 So, this is an efficiency rule. It's
21 sounder and it's more efficient.
22 The most -- Dr. Powers hinted at the most
23 far reaching aspect of this rule, is we're going to
24 propose a non-expiring license for research reactors.
25 And, I'll let Al, Duane and Bob get into that.

1 But, I'll mention that there are eight
2 other aspects to the rule.
3 We'll start out by talking about the non-
4 power production or utilization facility and the
5 definition that we're proposing for that.
6 We're going to define license renewal for
7 testing facility and NPUFs under 50.22.
8 We want to require all NPUF licensees to
9 submit routine FSARs, very good practice, sound
10 housekeeping.
11 We want to amend the current timely
12 renewal provision which is anything but timely.
13 We want to extend the -- we want to
14 provide an accident dose criterion for NPUFs that's
15 less restrictive than the 10 CFR 20 that we currently
16 use.
17 We want to extend the applicability of
18 50.59 to NPUFs regardless of decommissioning status.
19 And, we want to clarify requirements for
20 meeting the current environment report provisions of
21 51.45.
22 And, finally, we would also like NPUFs to
23 not have to submit the financial qualification
24 information for license renewal, and that's to achieve
25 parity with power reactors.

1 The staff presenters today are Al Adams,
2 who is the Branch Chief for Research and Test
3 Reactors, Duane Hardesty, who has been the technical
4 lead on this rulemaking and he is a PM in Al's branch
5 and Bob Beall, who has done a wonderful job PMing this
6 rule, he has the lead from the project management side
7 of the house.
8 And, with that, I'll turn it back to Dr.
9 Powers.
10 CHAIRMAN BLEY: Excuse me, Dana.
11 For those listening in and here in the
12 room, you heard an acronym, NPUF, which is new.
13 You've probably figured out that that's non-power
14 production and utilization facility.
15 MS. GAVRILAS: Thank you.
16 MEMBER POWERS: Which will get defined in
17 detail as the presentation goes on.
18 I just want to interject a couple of
19 points that I'm sure you're going to touch on, but
20 just to emphasize.
21 One is, indeed, this five-year periodic
22 updating to the FSAR which is a vast step forward, if
23 you would ask me, to the updating the FSAR every time
24 you get a license rule every 20 years.
25 And, that the -- though the proposed dose

criterion is different than the 10 CFR Part 20, it's also different than the 10 CFR Part 100. And, a good deal more risk criterion than the 10 CFR Part 100.

With that, I think we're in a position to progress on the presentations. And, Duane, I am told by those who claim to know, that you are going to lead us through this presentation.

MR. HARDESTY: Okay. So, Mirela already discussed who we are, so we'll move right on into the purpose of the NPUF proposed rule.

The Commission directed the staff to streamline the license renewal process due to a backlog of renewal applications that were caused by several factors. These factors have been described in the SECY papers that were exchanged with the Commission as well as in our Federal Register Notice that we currently published.

I can go into them if anyone would like to know more details, however, I would propose that given our time limits, that I would just briefly --

MEMBER POWERS: It is very well written up in the written material associated with this.

MR. HARDESTY: Thank you.

MEMBER POWERS: Well, I would say in general, the written material is very clear. I mean,

1 it's not confusing and things like that.

2 MR. HARDESTY: That's good to hear.

3 MEMBER POWERS: Of course, it does raise

4 the issue of the queuing theory that exists on

5 regulations within the Agency because it's a

6 confounding of episodic events that divert the staff

7 away from these things if it comes up.

8 I agree with you, let's charge on without

9 recounting past history.

10 MR. HARDESTY: Thank you, sir.

11 And, just for some background, the

12 regulatory basis that supported or justified

13 rulemaking was completed in August of 2012.

14 So, when we were at the Subcommittee

15 meeting, one of the members, I believe it was Mr.

16 Bley, had asked for a graphic that would better depict

17 the facilities that were impacted by this rulemaking.

18 And so, this was our collaborative effort

19 to bring something together that showed the

20 relationship of the NPUF entities.

21 So, under Section 101 of the Atomic Energy

22 Act as amended, we have the authority to license

23 production and utilization facilities.

24 The non-power production utilization

25 facilities themselves are provided for in Section 103

1 and Section 104 of the Atomic Energy Act.

2 At the bottom there, you'll see we were

3 very careful to name it commercial medical

4 radioisotope irradiation and production facilities.

5 That is the -- this whole graph is kind of the state

6 of being as it is or will be in the very near future.

7 So, that captures the new medical isotope

8 facilities that are coming in as Class 103 which is a

9 commercial entity.

10 They do share some commonality with the

11 research reactors and testing facilities that have

12 longstanding license in our group. They share

13 research reactor characteristics and that the --

14 potentially, they could do some research related to

15 medical isotope as do the research reactors.

16 And then, they also share some

17 characteristics with the testing facilities.

18 At the top there in the Class 104 A or C,

19 I would be careful to note, although the graph doesn't

20 really explain it well, that you can have a Class 104

21 A or a Class 104 C research reactor.

22 However, you can only have a Class 104 C

23 testing facility by our regulations.

24 And, the significant difference between

25 them, while you see that they share a common research

1 mission, is that they are defined by their risk.

2 And so, we have -- I'm sorry, by their

3 notable safety considerations. And, the notable

4 safety considerations, again, are defined in the

5 regulations under the definition for a testing

6 facility.

7 And, what they provide for is a power

8 level that if the testing facility is greater than 10

9 megawatts or the facility is greater than 10

10 megawatts, it becomes a testing facility licensed

11 under 104 C but subject to the other criteria for a

12 testing facility, most notably Part 100 as Dr. Powers

13 already mentioned.

14 The other criteria is great than megawatt

15 with these notable safety considerations which is what

16 we cause as. And, you can see in the bottom corner of

17 the slide that those are circulating loop through the

18 core used for fuel experiments, a liquid fuel loading

19 or there's a large experimental facility and it's

20 specifically defined as being greater than 16 square

21 inches in a cross section.

22 So, that is basically the difference on

23 whether or not they're going to be a research reactor

24 or a testing facility.

25 The bulk of our facilities fall into the

1 research reactor category.

2 There are currently 30 research reactors,

3 one testing facility and two commercial medical

4 radioisotope production facilities that are being

5 considered for licensing.

6 And, with that, I will pass it over to Mr.

7 Beall.

8 MR. BEALL: Good morning.

9 What I'm going to do today is talk about

10 the nine proposed changes that this rulemaking is

11 addressing.

12 So, the first one is definition of an

13 NPUF. So, this proposed change would address

14 inconsistencies in definitions and terminology

15 associated with NPUF licensed under Sections 10 CFR

16 50.22 and 10 CFR 50.21(a) and ©.

17 The current definitions for NPUF

18 facilities also do not adequately cover the proposed

19 medical isotope irradiation and processing facilities

20 that Duane talked about.

21 The NRC is proposing to add a specific

22 definition for non-power production or utilization

23 facilities to 10 CFR 50.2 to establish a term that is

24 flexible enough to capture all non-power facilities or

25 NPUFs under Part 50.

1 Proposed change two is licensing terms.
2 The Atomic Energy Act currently does not establish a
3 licensing term for Class 104 A and 104 C facilities.
4 Currently, the NRC licenses those
5 facilities under 10 CFR 50.51(a) for a term less than
6 equal to 40 years.
7 The staff currently licenses those NPUF
8 facilities for approximately 20 year terms for both
9 renewals and initial licensees.
10 The non-expiring licenses are consistent
11 with the Atomic Energy Act Section 104 which states
12 that imposing only such minimal amount of regulations
13 under this Act to promote the common defense and
14 security and to protect health and safety of the
15 public.
16 As such, the proposed action will reduce
17 the burden on both the licensees and NRC staff.
18 MEMBER POWERS: The Act is very explicit
19 in distinguishing these facilities from power product
20 facilities with respect to regulation on this point?
21 MR. BEALL: Yes, sir.
22 MEMBER POWERS: It's more emphasized in
23 your written documents than it is on this slide. That
24 there's this distinguishing language in the Act
25 itself.

1 MR. BEALL: Right.

2 MR. ADAMS: And, if I can make a comment
3 that for these -- for this type of reactor facility,
4 the Act also requires what's called minimum
5 regulation. And, we believe that this is an, you
6 know, was a purposeful example of minimum regulation.

7 So, I think, you know, the most
8 significant change out of the changes that we are
9 suggesting is the non-expiring license and the next
10 slides, I'd like to explain a little bit of the
11 reasoning that led us to this place.

12 So, I want to start by discussing the
13 design and operational characteristics of research
14 reactors that result in no notable safety
15 considerations.

16 So, this discussion is applicable to the
17 NPUF type of the research reactors. It's not related
18 to testing facilities or commercial medical isotope
19 facilities who will be subject to license renewal.

20 So, research reactors operate at low power
21 levels, currently 5 watts to 10 megawatts thermal
22 power.

23 Most research reactors operate
24 periodically. Your normal research reactor will run
25 several days a week for several hours to support

education, training.

Because of this, research reactors have a small inventory of fission product. The maximum hypothetical fission product release accident for all currently operating research reactors results in occupational and public doses with in 10 CFR Part 20 limits.

Compared to power reactors, these are low energy systems. They operate at low temperatures and low pressures.

For example, most reactor pools are less than 40 degrees C during operation. There's only one pressurized research reactor and it operates at less than a 100 pounds.

Limited components are in what I would consider a harsh radiation environment.

Because of the power level and operating history, most reactors produce minimal decay heat. In general, facilities with a power level below 2 megawatts thermal can be air cooled in the event of loss of coolant.

Some facilities over 2 megawatts have emergency core cooling systems that may need to operate for a limited period of time after an accident.

1 This is dependent on operating history,
2 operating power and how long after shutdown that a
3 loss of cooling accident would occur.
4 These ECCS systems are subject to regular
5 surveillance requirements to help ensure operability
6 if needed.
7 MEMBER POWERS: There is in the discussion
8 of this frequent mention of risk. And, those mentions
9 make it apparent that this is qualitative risk, human
10 judgment on what risk is.
11 Have there ever been any quantitative risk
12 assessments particularly of these low power systems
13 with the -- the ones with the ECCS system?
14 MR. ADAMS: The answer to your question is
15 no. We know that, you know, out in the academic
16 world, there have been a few exercises that have been
17 done to support either theses or dissertations.
18 I looked at this a number of years ago and
19 I looked at facilities where this has been done within
20 the DOE world where you're talking about 100 megawatt
21 systems and overseas, there's been a few examples.
22 For example, in Australia, for the new
23 OPAL reactor, there was some work done as part of the
24 design process and that resulted in some design
25 changes which decreased the probability of uncovering

1 the core.

2 So, within our world, no, there's nothing

3 that's been done within NRC space in this area.

4 MEMBER POWERS: Particularly for those

5 systems that do require an ECCS, ones that can be air

6 cooled, of course, have an advantage. But, those that

7 do require a short period of operation of the ECCS,

8 presumably, they are subject to roughly the same kind

9 of seismic hazard as any nearby power reactor.

10 MR. ADAMS: And, these, you know, these

11 tend to be the higher powered research reactors. And,

12 as you go up in power level, we see that the lower

13 power facilities, and you know, facilities that were,

14 you know, primarily built in the '50s and '60s, that

15 they were built to the seismic building codes for that

16 location.

17 Or, in a lot of cases, we see, you know,

18 the next level up from the codes.

19 When you get to these higher power

20 facilities, you know, University of Missouri, MIT,

21 they were designed to specific accelerations, seismic

22 accelerations, including ECCS systems.

23 There's, you know, testing to make sure

24 these systems can do what they need to do. And, you

25 know, we've, you know, we've never had an actual

actuation of a system for safety purposes.

Once in a while there, you know, over history one has turned on and, you know, put water into the system which has its own -- it creates its own excitement.

MEMBER POWERS: And, presumably, they're subject to the same station blackout frequencies?

MR. ADAMS: They tend to be passive systems. For example, the reactor I worked at had one of these systems and, indeed, if you lost power and the emergency generator, it didn't pick up the load properly, you were putting, you know, water into the pool.

So, they tend to be systems that are failsafe.

MEMBER SKILLMAN: Al, I think we've talked about this back in the Subcommittee meeting, but perhaps you could refresh our memory.

When you describe these features of these reactors, it's easy to get the feeling, hey, nothing bad can really happen, no matter what you do. These things are kind of always safe all the time. Low power level, not using it a whole lot, not a whole lot of risk, particularly to the public.

What prevents there from being an attitude

of complacency on these machines?

MR. ADAMS: I think that, and that's something that the NRC looks at, the inspection program looks at. And, indeed, you know, over history, you know, I think we've seen some facilities that have drifted in that direction and it starts manifesting itself in problems that the inspection program that the staff does pick up upon and makes the licensee correct.

So, it is something that we look out for and, indeed, it is something that, you know, over history, we have seen.

MEMBER SKILLMAN: And, how often would those inspections occur so as to catch that complacency?

MR. ADAMS: Sure. The inspection program normally under the routine inspections, inspectors are on site at least twice a year. More often if an event or performance dictates that.

MEMBER SKILLMAN: Okay, thank you, Al.

MR. ADAMS: So, because we have a mild operating environment, a simple design that the staff has identified no notable aging issues.

Technical specifications surveillance requirements help to ensure the operability of systems

1 and components.

2 The components the staff has historically

3 focused on for aging are fuel cladding and

4 instrumentation and control systems.

5 Fuel failures are very rare. Over the

6 last 30 years, there have been a few small clad

7 failures that were quickly identified by licensees.

8 The fuel removed from service with no impact on public

9 health and safety.

10 The failures we've seen in I&C systems

11 normally lead to reactor shutdown.

12 Systems are normally upgraded by licensees

13 due to either inability to require replacement parts

14 or desire to upgrade technology.

15 One are where we have seen aging is in the

16 reactor coolant boundary. There have been several

17 pool failures over the years of both concrete pools

18 and what I call lined pools, concrete pools that are

19 lined with either aluminum or stainless steel liners.

20 Because of this history, the staff now ask

21 licensees to evaluate coolant heat exchanger failure

22 in their SARs. So, this is including in the staff

23 safety analysis report as an analyzed occurrence.

24 The majority of research reactors have

25 coolant radioactivity levels within the Part 20 limits

1 to release to the environment or the sanitary store.
2 Tech specs require monitoring of pool
3 level, coolant chemistry and coolant radioactive
4 reactivity content.
5 And, licensees monitor water additions to
6 identify trends that can indicate a water loss. You
7 know, when you have an open pool system, we normally
8 ask licensees to tell us what's the minimum real loss
9 they can dictate, you know, given evaporation and, you
10 know, with humidity going up and down, you know,
11 normal leakage past pump seals and things like that.
12 So, that's a discussion we have during the current
13 round of license renewals with licensees to determine
14 what level of leakage they could identify.
15 Finally, the design basis of these
16 facilities evolve slowly over time. The NRC receives
17 approximately five license amendment requests each
18 year from all of these licensees combined.
19 Also, on average, each one of these
20 licensees reports about five 10 CFR 50.59 evaluations
21 per year.
22 Very few of the NRC rulemakings impact the
23 operation of a research reactor.
24 So, we've spent a lot of time considering
25 the nexus between license renewal and safety. So,

1 when the first power reactors are facing license
2 renewal, research reactor renewal was already an
3 established process.
4 We looked at the framework that was being
5 developed at the time for power reactor license
6 renewal which, as you know, focused on aging of
7 structure systems and components important to
8 continued safety.
9 We did not include research reactors in
10 the power reactor framework for two reasons. One, we
11 already had an established license renewal process
12 and, also, we couldn't envision what license renewal
13 would encompass for research reactors if limited to
14 aging issues.
15 So, the question is, without notable aging
16 issues, what does taking a snapshot of research
17 reactors once every 20 or 40 years contribute to
18 safety?
19 After a lot of thinking and discussion, we
20 came to the answer that nothing of safety importance.
21 But, there are other processes in place and the
22 actions we have taken or are proposing to take
23 contribute more to the continuing safety than
24 performing a license renewal every 20 years.
25 NUREG-1537 is our format and content

1 guidance for licensees and the standard review plan
2 for the staff for non-power facility licensing
3 actions. It was issued in 1996.
4 So, prior to 1996, there was no
5 comprehensive guidance for licensees or for the staff.
6 All of the facilities will undergo a
7 license renewal review using NUREG-1537 before
8 entering into the non-expiring license process. This
9 ensures a comprehensive and consistent licensing basis
10 using established guidance for the licensees.
11 And, for the staff, we have a licensing
12 basis that was reviewed and documented in the safety
13 evaluation report using a standard review plan.
14 So, we have solid documented licensing
15 basis of the safety of these facilities that will be
16 entering into a non-expiring license process.
17 The inspection program is comprehensive
18 looking at all aspects of facility operations. As I
19 mentioned, inspectors are on site several -- at least
20 several times a year. Any deterioration of a
21 licensing performance will be discovered and corrected
22 and the inspection results are reviewed for adverse
23 trends that could indicate new aging issues.
24 The tech specs contain surveillance
25 requirements to help ensure safety limits, limiting

1 safety system settings and limiting conditions for
2 operation are met.
3 The administrative requirements in the
4 tech specs require safety committee review of
5 potentially significant activities and audits of
6 reactor operations and events. So, negative trends
7 should be identified by licensees.
8 The tech specs also contain reporting
9 requirements in the form of annual reports and event
10 reporting and I'll give you a few examples of what
11 these requirements are.
12 For example, the technical specifications
13 require licensees in their annual report to tell us a
14 number of things, but two of the things they tell us
15 is the number of unplanned shutdowns and inadvertent
16 SCRAMs including the reasons therefore and corrective
17 actions taken to reduce a recurrence.
18 And, I'm reading an actual tech spec for
19 that. That comes out of the -- that all research
20 reactors have in accordance with the ANS Standard on
21 Technical Specifications.
22 Another thing we learned from the annual
23 reports is a tabulation of major preventative and
24 corrective maintenance operations having safety
25 significance. So, we get a window on what components

1 are breaking, what type of maintenance is being done.
2 And, again, what we're looking for is trends that
3 would indicate something, you know, something new
4 happening.
5 The reportable events are items that must
6 be reported to NRC within 14 days, as you find in the
7 tech specs.
8 And, two examples of them would include a
9 required reactor safety component malfunction that
10 renders or could render the safety system incapable of
11 performing its intended safety function.
12 Another example is abnormal or significant
13 degradation in reactor fuel or cladding or both,
14 coolant boundary or confined boundary.
15 So, because of these requirements, the
16 staff will be aware of any negative trends and
17 facility performance much sooner than the 20 years in
18 between renewal periods and the staff will take
19 appropriate actions if safety becomes an issue.
20 MEMBER SKILLMAN: Al, are there any events
21 that require an immediate report? You just mentioned
22 --
23 MR. ADAMS: Yes, so the --
24 MEMBER SKILLMAN: -- 14 day reports.
25 MR. ADAMS: Right, so the licensees are

still subject to all the requirements of the regulations for reporting. And, next day reporting is required for a violation of the safety limit. And, if that happens, the facility has to shutdown, report to the NRC no later than the next day.

Facility management and the safety committee has to be notified. An investigation has to be done which is submitted to NRC and the facility can't restart without getting approval from NRC.

MEMBER SKILLMAN: Okay, thanks, Al.

MR. ADAMS: So, these measures -- finally, these measures will be strengthened by the proposed rule requirements, in particular, requirement to submit updates to the SAR at a greater frequency than the 20 to 40 year interval that we see SAR updates now.

The staff believes that the safety of research reactor design and operation along with the measures I've discussed and the proposed rule will ensure an acceptable level of continuing safety without the need for a license renewal.

And, with that, I'll turn back over to Bob.

MR. BEALL: If the -- the third proposed change for this rulemaking has to do with the license

1 renewal process.

2 The staff is proposing to create a new

3 Section 10 CFR 51.35 that would consolidate into one

4 section the existing regulatory requirements for

5 license renewal for current and future NPUF licensees

6 licensed under Section 50.22 and testing facilities.

7 The proposed rule would not impose any new

8 regulations on these facilities, but the staff would

9 also make conforming changes to Section 50.8 to

10 reflect the improved information collecting

11 requirements as proposed in the new Section 51.35.

12 The fourth proposed change has to do with

13 the updating of the FSARs.

14 The staff found that licensees were not

15 always able to provide in their license renewal

16 applications documentation describing in details their

17 licensing basis.

18 Some licensees also had difficulty

19 documenting the necessary updates to a licensing

20 basis.

21 The current application and review process

22 is overly burdensome for both licensee and the staff

23 because of incomplete information supplied in some of

24 the license renewal applications.

25 The proposed rule would require all NPUF

1 licensees to submit routine FSAR updates to the NRC
2 every five years according to the changes in Sections
3 10 CFR 50.71(e).
4 By requiring periodic updates to the FSAR,
5 the staff anticipates that licensees would document
6 changes in the licensing basis in a more timely
7 manner.
8 The staff anticipates that the changes
9 would result in minimal additional burden for the
10 licensees and the staff, largely because the licensees
11 are currently required by 50.59 to keep the FSARs up
12 to date.
13 MEMBER POWERS: What -- it was not unclear
14 to me once you hit steady state and there's a
15 transition regime we have to go through to get into
16 this, what the licensee would submit on this five year
17 interval, would it be just change pages or would they
18 submit a whole FSAR changes as appropriate?
19 MR. HARDESTY: So, this is Duane Hardesty.
20 The regulation change that we're making to
21 50.71(e) mirrors what the power reactors have done in
22 the past. So, they have the option of either, under
23 50.4, communications, submitting changed pages with
24 indications by change bars that show us what has
25 changed as well as submitting a full FSAR.

1 We developed a Reg Guide 22 -- well, it's
2 a draft Reg Guide that provides guidance on what they
3 should provide in that update. We actually asked that
4 -- we kind of point them in the direction of
5 submitting a full FSAR simply for the fact that it'll
6 be much less cumbersome on everybody as well as --
7 MEMBER POWERS: Well, there'll be the
8 isolated guy that has -- corrects two spelling errors
9 and things like that but it's unnecessary. But,
10 anybody that does more draconian stuff, you would want
11 him to submit a full one. So, you're providing the
12 option and you're pleading with them to be rational in
13 choosing between those options.
14 MR. HARDESTY: We're also hoping to drive
15 them in the format of electronic submissions, although
16 we give them option to repay those --
17 MEMBER POWERS: Oh, do you?
18 MR. HARDESTY: Yes, sir.
19 MEMBER POWERS: I thought -- I mean, you
20 allow the option of being electronic or paper? I
21 didn't realize that.
22 MR. BEALL: Also, the FSAR updates are
23 important, as Al mentioned, about the NRC inspection
24 program and for the effective licensee operator
25 training and examinations.

1 Also, updated FSARs will increase the
2 timeliness of the NRC's review of future license
3 renewal applications. This is because we anticipate
4 fewer number of requests for additional information on
5 the renewal applications.
6 The fifth proposed change has to do with
7 timely renewal provisions.
8 The current regulations allow NPUF
9 licensees to submit license renewal applications as
10 late as 30 days before the expiration of their
11 existing licenses.
12 The 30 days is not sufficient time period
13 for the staff to adequately assess the license renewal
14 application. As a result, the staff has anticipated
15 acceptance of license renewal applications and address
16 their deficiencies during the license renewal process
17 largely through submitting requests for additional
18 information to the licensee.
19 This increases both the burden on the
20 licensee and the staff. The staff is proposing
21 revisions to the timely renewal provisions to Class
22 103 and testing facilities to a length of time for the
23 staff to review the sufficiency of the license renewal
24 application before the current license expires.
25 The proposed rulemaking would allow a

1 licensee under Sections 50.22, or a testing facility,
2 to file a license renewal application at least two
3 years before the expiration of their existing license.
4 The existing licenses will not be deemed
5 to have expired until the application has been finally
6 determined by the Commission.
7 The proposed rulemaking will ensure that
8 the staff has adequate time to review the license
9 renewal application while the facility continues to
10 operate under the terms of its current license.
11 The staff is also proposing to eliminate
12 this requirement for facilities other than testing
13 facilities licensed under Class 104 A and C as these
14 licensees will no long have a license expiration date.
15 That's the proposed change two we just talked about
16 earlier.
17 Proposed change number six, accident dose
18 criterion. Currently, there is a no accident dose
19 criteria for the NRC regulations to determine the
20 acceptability of licensing research reactors and
21 medical isotope irradiation in processing facilities.
22 Testing facilities apply the dose accident
23 criteria in Part 100. Accident dose criteria are not
24 dose limits, that is, that they do not define
25 acceptable dose but are used as reference or numeral

values to ensure that, in the unlikely event of an accident, the public will not suffer undue risk to health effects.

In May of 1972, the Atomic Safety and License Appeal Board suggested that the standards of Part 20 are unduly restrictive as accident dose criteria for research reactors.

The NRC had generally found that pre-1990, public dose limits of 0.5 rem for Part 20 to be acceptable for accident criteria.

In 1990, the NRC amended Part 20 and lowered the public dose limit to 0.1 rem.

Because of NPUFs low potential radiological risk to the environment and the public, the 0.1 rem dose limit is unnecessarily restrictive as applied to accident consequences such as the maximum hypothetical dose considered in NPUF license renewal applications.

In 1992, the Environmental Protection Agency, or EPA, published Protection Action Guides, or PAGs, in its Manual of Protection Action Guides and Protective Actions for Nuclear Accidents.

The EPA published a revision to the PAG Manuals in 2013. The PAGs with dose guidelines to support decisions that trigger protective actions such

as staying indoors or evacuation to protect the public during radiological incidents.

In the early phases of a radiological incident, the trigger for the protective actions are shelter in place or evacuation of the public ranges from 1 to 5 rem.

Below 1 rem, there is no protective actions initiated to avoid the protected dose guideline. Therefore, the proposed action dose criterion of 1 rem provides adequate protection of the public from unnecessary exposure to radiation in the unlikely event of an accident.

Proposed change seven has to do with the applicability of 10 CFR 50.59.

For NPUFs licensees that had fuel removed from the site, the staff is required to add a license condition to allow the licensee to make changes to their facility or changes to the procedures per 10 CFR 50.59.

The license amendment process imposes administrative burden on the licensees and NRC staff which could be eliminated with the proposed regulatory change.

The proposed rule will revise the wording of 50.59(b) which currently does not apply to

1 requirements of 50.59 to NPUFs whose licenses have
2 been amended to cease operation and have returned all
3 of their fuel to the Department of Energy.
4 The staff is proposing to extend the
5 applicability of 50.59 to NPUFs regardless of their
6 decommissioning status.
7 Proposed change eight has to do with
8 environmental reporting requirements. The NRC
9 environmental protection regulations under 10 CFR Part
10 51 implement the environmental -- excuse me --
11 National Environmental Policy Act, or NEPA, of 1969.
12 In accordance with NEPA and 10 CFR Part
13 51, the staff is required to review and evaluate the
14 potential environmental impacts of the proposed
15 licensing and regulatory actions and to prepare the
16 appropriate environmental review documents such as an
17 environmental assessment or an environmental impact
18 statement.
19 To assist the staff in meeting its
20 environmental review requirements, the staff has
21 historically relied on Sections 10 CFR 51.41 and the
22 guidance in NUREG-1737 to request pertinent
23 environmental information from NPUF applicants.
24 The proposed rule will specifically
25 require NPUF applicants to submit an environmental

1 report consistent with the existing requirements of
2 the 10 CFR 51.45.
3 The proposed requirements will provide
4 clarify for NPUF applicants and improve the efficiency
5 of the NRC staff in meeting its environmental review
6 requirements.
7 The last proposed change is number nine.
8 It has to do with financial qualifications.
9 Over the last few decades, the Commission
10 has assessed that the nexus between financial and
11 safety is at best indirect and there's little, if any,
12 doubt to support the statement of the contrary.
13 The financial qualification requirements
14 in 10 CFR 50.33 requires the licensee to provide
15 evidence that they have reasonable assurance of
16 obtaining funds necessary to cover estimated operation
17 costs for the first five years of operation.
18 Currently, this requires licensees at the
19 time of initial licensing and license renewal to
20 provide financial statements and forecasts for future
21 operating costs.
22 Licensees will still need to provide
23 financial statements and forecasts of evidence in
24 costs at initial licensing.
25 There's also another rulemaking ongoing

1 that is reassessing the financial requirements at the
2 time of initial licensing for all 10 CFR Part 50
3 licensees including NPUFs.
4 And, the Commission provided the staff
5 direction on this rulemaking in SRM SECY-13-0124.
6 Also, the Commission does maintain broad
7 authority through its regulations to request
8 additional financial information from a licensee
9 should it have just cause to do so.
10 As such, the elimination of the financial
11 qualification requirements at license renewal is
12 consistent with the 2004 rulemaking which eliminated
13 the financial qualification requirements at license
14 renewal for power reactors.
15 As part of all rulemaking, the staff also
16 did a regulatory analysis and looked at the backfit
17 considerations for this proposed rulemaking.
18 As a result of the proposed rule and
19 implementing guidelines, the staff estimated that the
20 NPUF licensees will incur a one-time implementation
21 cost of $140,000.00 followed by a total operation cost
22 of $1.6 million over the 20 year analyst period.
23 The proposed rule implementing guidance
24 will result in a total one-time cost to the NRC of
25 $720,000.00 to complete the proposed and final

1 rulemaking and oversee the implementation of the new
2 NPUF license renewal requirements.
3 This one-time cost will be followed by a
4 total operation cost of approximately $1.8 million
5 over 20 years for the payer to be re-analyzed.
6 In terms of the quantitative benefits
7 associated with this proposed rulemaking, NPUFs and
8 NRC will receive incremental benefits from the
9 elimination of the license renewal.
10 For NPUFs, the proposed rulemaking in
11 total will result in $5.5 million in cost savings over
12 20 years. And, for the NRC, the proposed rulemaking
13 in total will result in a $12 million in total cost
14 savings over 20 years.
15 Rather than taking no action, the staff
16 concludes that the quantitative benefits and costs of
17 this proposed rulemaking would address the
18 inefficiencies and existing licensing issues affecting
19 the NPUF license renewal process.
20 The proposed rulemaking results in the
21 benefit associated with increased regulatory
22 efficiency and as well as having benefits to the
23 public health and safety. And, with a total net
24 benefit of $13 million for the NRC and the licensees.
25 The staff also looked at, in the

1 regulatory basis, Section 10 CFR 51.09 backfitting.
2 10 CFR 51.09 backfitting was expressed
3 solely in terms of nuclear power reactors. The
4 proposed definition of NPUFs would include non-power
5 reactors, testing facilities and other non-power
6 production utilization facilities.
7 Because Section 51.09 does not apply to
8 NPUFs, and this proposed rulemaking would apply to
9 NPUFs, the staff did not apply Section 51.09 to this
10 proposed rulemaking.
11 MEMBER REMPE: Robert, before you go too
12 far ahead --
13 MR. BEALL: Yes, ma'am?
14 MEMBER REMPE: On slide 15, you talked
15 about the financial qualification information.
16 MR. BEALL: Yes, ma'am?
17 MEMBER REMPE: And, you indicated the
18 staff does have the means to request financial
19 information if they have just cause. Has that means
20 ever been exercised and what is -- could you elaborate
21 on what just cause is?
22 MR. HARDESTY: Members from the financial
23 branch can address that.
24 MR. BOWERS: Hello, name is Tony Bowers,
25 I'm the Branch Chief for the Financial Analysis

1 International Projects Branch in NRR.

2 I don't know all the specifics, but I do

3 -- I am aware that the Commission's authority has been

4 invoked once before.

5 The regulation is 10 CFR 50.33(f)(5) and

6 I think the recent recession that we experienced,

7 there were some questions about licensees financial

8 standing and, in particular, Exelon. And, the

9 Commission, through the staff, actually asked for

10 additional financial information.

11 The staff has developed Interim Staff

12 Guidance for how to actually solicit that information

13 from its licensees.

14 MEMBER REMPE: Okay, thank you.

15 MR. BEALL: The NPUF proposed rule package

16 consists of a SECY paper, Federal Register Notice,

17 regulatory analysis and backfit discussions, we have

18 environmental assessment report. As Duane mentioned,

19 there is also a draft Reg Guide and that's DG-2006.

20 Right now, the current status of that

21 package is it's being reviewed by OGC for their no

22 legal objection review. And, that should be due -- it

23 should be completed some time by the late this

24 weekend, excuse me, end of this week.

25 Also, we've also created notification

letters for Congress notifying them of this proposed rulemaking. And, we have an OMB statement that we'll send this information, this package, to OMB for their review also.

And lastly, we have some internal documents that are being revised also. The -- in DPR there's a Project Manager's Handbook and a couple of sections in that document will be updated to reflect that changes in this rulemaking.

Our current schedule is that we are on schedule to have the package to the Commission by April 1st. Then the Commission will review the package and hopefully send us an SRM directing us to publish the proposed rule and all the supporting documents. They will all go out at the same time for a 75 day comment period.

The staff will also hold a public meeting during the comment period including one during the TRTR annual conference in Mexico in August time frame.

And, we should have a final rule issued early in 2018.

MEMBER POWERS: Let me ask as a favor here that you let us know when you're having these public meetings?

MR. BEALL: Sure can. We'll do that.

1 MR. HARDESTY: This is Duane Hardesty.
2 We'll actually have a webinar, too, so I
3 can give you all that information.
4 MEMBER POWERS: Yes, just try to keep us
5 informed on these because in this particular regime,
6 these public meetings might be fairly important,
7 illuminating, et cetera.
8 MR. BEALL: Yes, we'll also come back to
9 the Committee with the final rule.
10 MEMBER POWERS: Yes.
11 MR. BEALL: The Subcommittee and the Full
12 Committee, too. So, you'll definitely be seeing us
13 again.
14 MEMBER POWERS: I'm just anticipating it
15 might save everybody a lot of work if we peered in on
16 at least one of these public meetings to get a feel of
17 where the things are less clear or other things might
18 be done, just informational type is all it is.
19 MR. ADAMS: And, I'll point out that we
20 have been discussing these concepts with the
21 community. We've had I think three public meetings,
22 four, we've had four public meetings with the
23 licensees.
24 MEMBER POWERS: Yes, you indicated that in
25 the writeup and I said, darn, I wish I had attended at

1 least one of those to just to get a feel for the
2 general tenor or things.
3 MR. HARDESTY: Yes, you missed out on
4 Oregon, Boston, San Francisco, or San Diego rather.
5 MEMBER POWERS: So far, you haven't listed
6 anything I've missed.
7 MEMBER REMPE: But, since we missed them,
8 what type of comments did you receive? I'm curious.
9 MR. HARDESTY: The comments we received
10 have varied greatly as we've come through the process
11 largely because we -- when we first started, we were
12 gelling what we wanted to do and we were soliciting a
13 lot of input from the licensees and the stakeholders
14 on how they felt.
15 So, as we got more centralized into what
16 we did, the last public meeting, we had -- there was
17 a process that was included called a licensing basis
18 assessment which has been discarded now as not being
19 an efficient and effective way to implement the
20 streamlining.
21 But, that was one of the things that the
22 licensees, I think, in general, did not like because
23 they felt like it was a five year license renewal as
24 opposed to a 20 year license renewal.
25 And, we came to the epiphany that they

1 were not needed because of everything we had and it
2 wasn't going to add any benefit. So, we have since
3 eliminated that.
4 Everything else that's left in the package
5 that we did not get any negative feedback from, we are
6 adding a couple of requirements, obviously, the
7 updates and I think that they have all come through
8 the license renewal process or are in the process of
9 being renewed and that has been so burdensome, in
10 general, I would say that they are looking forward to
11 that.
12 We actually have the executive chair for
13 the test training and research reactor group -- I'm
14 sorry --
15 MR. ADAMS: National Organization --
16 MR. HARDESTY: National Organization
17 Testing and Research Reactors, Jerry Jenkins, if you'd
18 like to hear more from him.
19 MEMBER REMPE: No, I remember him talking
20 at the Subcommittee meeting. But, others may want to
21 hear from him.
22 MR. BEALL: So, in summary, the proposed
23 rule will amend the regulations related to license
24 renewal for NPUFs licensed under sections of the
25 Atomic Energy Act 103 and 104 A and C.

1 There are nine proposed changes to the
2 regulations and with some major changes are most NPUFs
3 would have non-expiring licenses, Class 103 NPUFs and
4 testing facilities would continue operating under
5 existing licenses past their expiration date if the
6 licensee submits the license renewal applications at
7 least two years before the current license expires,
8 require NPUF licensees to submit an updated FSAR every
9 five years and we establish a dose accident criterion
10 for most NPUFs.
11 VICE CHAIRMAN CORRADINI: Can you remind
12 me, back on slide 19, I really like the summary
13 because, at least in the Subcommittee, I couldn't
14 remember what applied to what so that's very helpful.
15 But, remind me, I think I know why, but
16 remind me on number six for test reactors, that's a
17 case by case depending upon the test reactors or is
18 that just the power side?
19 MR. ADAMS: No, it's test reactors already
20 have an accident dose criterion and that's Part 100.
21 VICE CHAIRMAN CORRADINI: Okay. And which
22 is similar to Class 103, I assume?
23 MR. ADAMS: It's the same regulations that
24 apply to power plants.
25 VICE CHAIRMAN CORRADINI: Okay. All

1 right, thank you.

2 MR. BEALL: All right.

3 So, in conclusion, the proposed rule would

4 be eliminating license terms which would reduce the

5 burden on both the licensees and NRC staff. The

6 change is allowed by the Atomic Energy Act. It also

7 is consistent with providing minimal regulations on

8 NPUF licensees.

9 We will continue our oversight and

10 inspections by the staff at the NRR.

11 And also, the improved FSAR documentation

12 would be a reduce in burden for both the licensees and

13 the NRC staff.

14 The net total quantitative benefit of the

15 proposed rules is $13 million, so there's a cost

16 benefit savings there.

17 And, the proposed rule change will

18 maintain the safe operation of the facility while

19 protecting the public health and safety.

20 MEMBER POWERS: One of the primary

21 motivations of this rulemaking effort was, in fact, to

22 have some sort of efficiency or smoothness in the

23 updating of the FSAR process.

24 And, so you have 31 facilities that are

25 going to be submitting, is there some sort of keying

1 or queuing of this so you don't every five years get
2 a spike of 31?
3 MR. HARDESTY: Yes, this is Duane
4 Hardesty.
5 We have binned the existing licensees
6 according to where we feel they are in the process.
7 We originally, under Commission direction, developed
8 an Interim Staff Guidance for streamlining license
9 renewal which was our short term plan.
10 That short term plan did a focused review
11 of the license renewal. And, as Al mentioned earlier,
12 that used the guidance, the standard review plan of
13 NUREG-1537.
14 So, the facilities that will complete
15 license renewal immediately proceeding when we
16 envision this rule will go into effect will have their
17 licensing basis completely documented and most up to
18 date. So, we will order them into a non-expiring
19 license almost immediately. We have -- do have some
20 implementation phase for that.
21 Then there are two other bins, ones that
22 have yet to go under the license renewal process and
23 we will complete them and then when their license
24 renewal is done, we will issue a non-expiring license.
25 So, they will be staggered.

And then, the final group, we're giving a much longer implementation period. They went under the NUREG guidance but they were early on in the phase. And so, there's some lessons that we have learned over the process. We'll make sure that those have all been incorporated into their licensing basis and then we'll finally order them into the non-expiring license with a longer implementation period so that they're staggered out for us as well as them.

MEMBER POWERS: Yes. So, I mean you don't want --

MR. HARDESTY: No, we don't want them all coming in at once.

MEMBER POWERS: Coming in on December the 31st and getting right back into the same crisis mode you had before.

Do any of the members have additional questions on this matter?

Does anyone in the audience care to make a comment?

At this point, we can request comments from people who are listening in on this and that takes a minute or two. And, it's irritating.

CHAIRMAN BLEY: We are getting the phone line open.

1 MEMBER POWERS: Thank you.

2 And, in our high technology fashion that

3 we have, the only way I can ascertain whether anybody

4 -- whether the phone line is indeed open or not, I am

5 told that the only way to detect is if somebody says

6 something.

7 So, if you're listening on the phone line,

8 whether you want to make a comment or not, could you

9 just say hello?

10 MR. LEWIS: Marvin Lewis.

11 MEMBER POWERS: Hey, Marvin.

12 MR. LEWIS: I'm in Philadelphia, member of

13 the public. Hello.

14 MEMBER POWERS: If there's anyone on the

15 phone line that would care to make a comment, now is

16 your opportunity.

17 I don't hear anybody rushing to comment on

18 this. So, I think I can turn it over to you, Mr.

19 Chairman.

20 CHAIRMAN BLEY: I think you've set a

21 record, Dr. Powers. We're finished very early.

22 MEMBER POWERS: I am not responsible. Mr.

23 Hardesty and Mr. Beall and Mr. Adams were all --

24 CHAIRMAN BLEY: I think it was that little

25 diagram they did up that made it go so quickly.

1 MEMBER POWERS: I believe, however, to the
2 contrary, it was, in fact, the summary table on 19
3 that was most illuminating.
4 CHAIRMAN BLEY: Well, I have -- before we
5 take a break, I have several announcements.
6 Our next session is really an internal
7 one, but somebody might be interested, I guess,
8 following our rules.
9 We will reconvene at 10:45 to look at the
10 safety research program.
11 Announcements about tomorrow, there's
12 concern that there might be a significant storm and
13 the Commission has just made a couple of announcements
14 that I will share and interpret for us.
15 If the office is open with unscheduled
16 leave or telework, we'll be here as expected normally,
17 but the meeting with the Commission, we have a meeting
18 with the Commission tomorrow, will be at 10:00 as
19 scheduled.
20 If there's either a two hour delay or a
21 three hour delay, the meeting with the Commission will
22 occur at 10:30 tomorrow.
23 And, if federal offices are closed, the
24 Commission meeting with us will be postponed.
25 We will meet to do our letter writing as

scheduled, our staff will not be required to be here following NRC rules, but we'll have to have at least one show up so our staff needs to use their judgment depending on conditions.

But, we'll meet and try to go ahead. If we can't, we'll meet Saturday and finish our letters.

At this time, we'll take a recess until 10:45.

(Whereupon, the above-entitled matter went off the record at 9:32 a.m. and resumed at 1:01 p.m.)

CHAIRMAN BLEY: We're back to the afternoon session.

Before I turn it over, I want to announce to the people on the phone line, the phone line is muted during this discussion. When NEI is up, if they need someone on the phone line, we'll open it again.

Please mute your phones because we were getting a lot of noise coming through before we did that.

At this time, I'll turn the meeting over to Mr. John Stetkar.

MEMBER STETKAR: Thank you, Mr. Chairman.

The topic of this afternoon's meeting is the staff's white paper. And, they'll educate us on where they're headed on this soon to be turned into a

1 Commission briefing paper. Greg will enlighten us on
2 that, too, I'm sure, regarding plans forward for
3 resolution of what I call the Group 2 of the Fukushima
4 Tier 2 and Tier 3 recommendations and Greg and company
5 will enlighten us on that.
6 Before we start, I believe that Joy Rempe
7 has something.
8 MEMBER REMPE: I do need to acknowledge
9 that I have to limit my participation on discussions
10 from the topics in this session because of a conflict
11 of interest.
12 MEMBER STETKAR: Okay, thank you.
13 And, with that, I'll turn over the
14 proceedings to Greg Bowman. Greg?
15 MR. BOWMAN: Thank you, John.
16 My name is Greg Bowman, I'm one of the
17 Branch Chief's in Japan Lessons Learned Division.
18 As you know, coming to resolution on the
19 open Tier 2 and Tier 3 recommendations has been a very
20 high priority for the staff over the last six months
21 and pretty longer than that.
22 Bill will discuss this in more detail
23 momentarily, but, we provided the Commission with a
24 paper with our proposals back in October of 2015, the
25 SECY-15-0137.

1 Our evaluations in that paper included
2 consideration of the pre-Fukushima activities and the
3 regulatory framework in place before, the safety
4 enhancements that we made since the accident and
5 Commission decisions recent that -- the two Commission
6 decisions that bear on our consideration of these
7 recommendations.
8 We've also benefitted greatly from the
9 interactions with the public, industry and with the
10 ACRS in coming up with our recommendations, both in
11 that paper and what we're going to present to you
12 today.
13 We recently received the Commission's SRM
14 on our paper. That SRM included approval of closure
15 of a number of the recommendations and also our plans
16 for completing our review of the remaining
17 recommendations.
18 As John mentioned, today we'll be
19 discussing three recommendations that were categorized
20 as Group 2 in SECY-15-0137 and those were
21 recommendations that, at the time, we believe we had
22 sufficient basis to close but that we felt there was
23 a need for additional interactions both with the ACRS
24 and with the public and industry before we could make
25 a final recommendation to the Commission.

1 And, we do owe a final paper with our
2 final recommendations associated with those
3 recommendations to the Commission at the end of this
4 month.
5 MEMBER STETKAR: Greg, let me just state
6 so we have it on the record that the ACRS did write a
7 letter regarding SECY-15 --
8 MR. BOWMAN: 0137.
9 MEMBER STETKAR: 0137, too many papers out
10 there.
11 MR. BOWMAN: Yes.
12 MEMBER STETKAR: And, we did, as a
13 Committee, agree to the way that you had grouped the
14 items. And, we did also agree for your plans to close
15 the Group 1 items. So, that there isn't any
16 misunderstanding on that.
17 MR. BOWMAN: Thank you, John.
18 So, as part of our presentation this
19 afternoon, we'll provide background on the
20 recommendations, we'll give an overview of the
21 assessment process we used and the technical details
22 of our final evaluations.
23 We'll also provide you with a summary of
24 the changes we've made in response to comments we got
25 from ACRS both at the Subcommittee meeting we had just

a couple of weeks ago and then back in October before the last paper came up.

Our overall objective of the initiative, looking at the recommendations, was to determine if any of them would result in a substantial safety enhancement that would justify imposing a new requirement under the backfit rule.

So, notwithstanding where we're, you know, our recommendations for these issues, there will be activities that continue on in many areas, many related areas, for a number of years.

Joe will touch a little -- touch some on the -- some of the activities associated with severe accident management guidelines, what their research activities and other things that will be going on for many, many years regardless of where we end up with these recommendations.

So, with that, I'll turn things over to Bill Reckley to kick of the presentation for us.

MR. RECKLEY: Okay, I'll go through the first few slides pretty quickly because Greg already touched on most of them.

Again, just as a highest level background after the Fukushima accident, the NRC identified a fairly large number of recommendations, some coming

1 from the Near Term Task Force, some coming from
2 interactions with the ACRS and some coming from the
3 staff.
4 We prioritized those recommendations into
5 three tiers, 1, 2 and 3 with Tier 1 being those that
6 were undertaken without delay. That was largely the
7 issue once we had the orders and the issuance of
8 50.54(f) letters requesting licensees to do plant walk
9 downs and to do reevaluations of seismic and flooding
10 hazards.
11 Tier 2 and Tier 3 were a mix of items that
12 involved either further study, analysis. Some of them
13 depended on the outcomes of Tier 1. So, they were
14 just given a little longer schedule to resolve.
15 As Greg mentioned, we're now at the point
16 where we think, based on the progress made on Tier 1
17 activities, including mitigating strategies, we're at
18 a point where we can make determinations on the Tier
19 2 and Tier 3 items.
20 MEMBER STETKAR: Bill, just for my own
21 recollection, the Tier -- at the time, Tier 2 and Tier
22 3 were allocated to those tiers for the reasons that
23 you've highlighted on this slide. It wasn't
24 necessarily that something was in Tier 2 because it
25 was judged on a fundamental basis less important than

1 something was in Tier 1 or vice versa.

2 MR. RECKLEY: As time went on, the

3 distinction between Tier 2 and Tier 3 is really

4 blurred.

5 MEMBER STETKAR: But, a lot of it was, as

6 you've highlighted here, it's resource considerations

7 or dependency on a sequential set of assessments.

8 MR. RECKLEY: Right.

9 MEMBER STETKAR: Okay, thanks.

10 MR. RECKLEY: So, after a few years and

11 some limited work on the majority of the Tier 2 and

12 Tier 3 items, we, again, we felt we were ready to

13 close out many. So, we issued SECY-15-0137 in October

14 of 2015.

15 We identified what had already been

16 acknowledged in many cases that some of the Tier 2 and

17 Tier 3 activities had actually been already addressed

18 by the Tier 1 activities.

19 And, examples of that are the spent fuel

20 pool cooling recommendations, some of the emergency

21 planning recommendations that were subsumed into the

22 mitigating strategies order and related rulemaking.

23 The paper then outlined resolution plans

24 for the remainder of the Tier 2 and Tier 3 items.

25 And, as everyone's already mentioned, we divided those

1 into groups.

2 One were those that could be closed

3 immediately.

4 Two were those that the staff thought,

5 based on the initial technical work that was done,

6 could be closed but we had not had public interactions

7 or specific interactions with the ACRS.

8 And, Group 3, there needed a little more

9 evaluation or work on developing an actual program

10 plan. And so, they'll be done by the end of 2016.

11 We're here to discuss the Group 2 items

12 which are those listed here, vents for containment

13 designs other than Mark I and Mark II, keeping in mind

14 that activities from Mark I and Mark II containments

15 were already taking in the issuance of order 13-109.

16 Recommendation six, Near Term Task Force

17 recommendation six, related to hydrogen control and

18 mitigation inside the primary containment and also in

19 adjacent structures like the reactor building.

20 And a recommendation that was identified

21 from interactions with the ACRS related to possible

22 enhancements to reactor and containment

23 instrumentation to better address beyond design basis

24 events.

25 As John mentioned, the ACRS did write a

letter on November 16th related to SECY-15-0137. And, as Greg mentioned, the SRM was dated February 8th and largely approved what the staff had laid out in the paper.

And, I had gone through this on the Subcommittee meeting but, it's worth --

MEMBER STETKAR: Bill, we have ample time. I must admit, I haven't -- if I read the SRM, I forgot it.

You said largely, did it --

MR. RECKLEY: It didn't technically differ. It gave us some directions to consider this or consider that and gave us schedules as to when we needed to respond.

MR. BOWMAN: So, one of the items in the SRM that you all will be involved in is they asked for the other external hazard review. There's, you know, looking at things other than seismic and flooding to come back to the Commission with an interim deliverable in May.

So, we were coming to meet --

MEMBER STETKAR: Yes, we have a Subcommittee set up and we're planning to write a letter on that in that time frame to support that.

MR. RECKLEY: So, as Greg had mentioned,

one thing to keep in mind as we go through this is that the primary question that the staff is trying to answer is whether or not possible plant changes or procedure changes could be justified in terms of the backfit process and the regulatory analysis process for generic items such as a rulemaking.

This plot taken from our regulatory analysis guidelines, NUREG/BR-0058, lays out the flow chart.

Basically, involved us doing an assessment of the value of a potential regulatory requirement and assessing whether it delivers a substantial safety improvement. And, for that, we use the guidance in the NRC safety goal policy statement as a consideration.

The process then involves, assuming that there is a substantial safety improvement or a potential one, whether the estimated value of the plant change would warrant the cost associated with it, a cost benefit assessment.

There are exceptions to this, for example, as marked in the box, if it's significant enough that the Commission were to decide it was needed for reasonable assurance of adequate protection, then we don't do the cost benefit assessment, we go right to

1 the imposition of the requirement.

2 And, for us, that was noteworthy. That

3 was the Commission's logic in the issuance of Order

4 EA-12-49 for mitigating strategies and the initial

5 Order for EWR events, EA-12-50.

6 The other note that's on this slide that

7 I'd like to remind people is, this is not an NRC

8 activity that we undertake just for us. This is the

9 way federal government agencies, including, you know,

10 other regulatory agencies pursue whether to issue new

11 regulations or requirements.

12 Executive Orders, guidance from OMB and

13 GAO lay out how cost benefit assessments are done and

14 considered in those processes.

15 So, as has been mentioned, following SECY-

16 15-0137, we're often put in the quandary of owing

17 Commission documents, in this case, we owe them

18 another SECY paper by the end of this month, and

19 having to try to schedule interactions with the public

20 and the ACRS.

21 And so, the way we've come to do that in

22 Fukushima activities has been to been issue white

23 papers which are largely drafts of the subsequent SECY

24 paper.

25 We did that in this case and issued a

1 white paper to support interactions with ACRS, first,
2 the Fukushima Subcommittee two weeks ago and we also
3 had a public meeting in early January to discuss the
4 white paper.
5 We tried in the white paper to keep the
6 focus on the regulatory decision as to whether there
7 were justifications to impose requirements for any of
8 the three areas that we've talked about.
9 As Greg mentioned, to reflect Commission
10 decisions that have been made over the last couple of
11 years, really, as we've evolved in our guidance for
12 these considerations, not really evolved, we've
13 basically have just reinforced that the guidance that
14 was applicable before Fukushima would also continue to
15 be used for making decisions in the post-Fukushima
16 environment.
17 Acknowledge the various studies and
18 international activities have gone on and have
19 identified enhancements and, to some degree, those
20 different ways of doing it have been reflected in how
21 we handle new reactors.
22 And, again, this is a pre-Fukushima
23 activity but new reactors in the U.S. have, through
24 the severe accident policy statement the Commission
25 adopted in the '80s had to have severe accident design

features that were not imposed on operating reactors.

We acknowledge that in some countries, measures have been taken such as the installation of engineered filters, the installation of hydrogen recombiners and that those safety improvements have been made in some cases.

When we assess them against the thresholds associated with NRC's regulatory policies, we continue to find that they don't meet the backfit test for imposing those kinds of requirements on the operating fleet.

So, to talk about the specifics, I'm going to talk about the containment vents for other containments and Joe Sebrowsky's going to talk about the enhanced instrumentation. And, I'm also going to talk about the hydrogen issues, since those are closely related, containment and hydrogen issues.

So, in the initial assessment in SECY-15-0137, we included a basis for our initial finding that was largely based on the existing body of work, the containment performance improvement program that followed Three Mile Island and involved a lot of testing and studies in the '80s, NUREG-1150, the risk analysis work and the more recent State of the Art Reactor Consequence Analysis, or SOARCA, activities.

1 We looked at the available technical
2 information that we generated for the Mark I and Mark
3 II containment studies.
4 As was mentioned, we looked at previous
5 Commission decisions including things like SECY-12-
6 110, the economic consequence paper and then, more
7 recently, the SECY-15-65 and the decision to not
8 include severe accident management guidelines in the
9 rulemaking for mitigation of beyond design basis
10 events.
11 We undertook then to do an evaluation of
12 each containment type looking at that body of work.
13 And, again, reached the initial conclusion that
14 further study is unlikely to demonstrate the need for
15 regulatory action.
16 MEMBER REMPE: So, before you leave that,
17 just for information -- yes, I think my mic's on,
18 here, there we go, now I have stereo again.
19 But, I appreciated during the Subcommittee
20 meeting, the question was asked about how did you look
21 at the PWRs and the staff analyses, the containment
22 integrity, and I know you guys said there's a large
23 number of documents.
24 And, I was interested in one of the
25 document like NUREG/CR-6909 and it talked about

thermal degradation as well as pressure testing.

And, but, I believe the question that we asked was what was done with the staff evaluations and what containment and fragility curve was assumed? Was it something from the old IPEs and then they said, well, the temperatures didn't get above a certain value? Because NUREG/CR-6906 I believe is the number --

MR. RECKLEY: 6906.

MEMBER REMPE: Yes, did talk about the temperature fix and I guess I couldn't detect from the large body of documents you sent me what was done actually for these analyses.

Did you use the old IPE fragility curves or did you use something that considered thermal as well as pressure degradation?

MR. RECKLEY: There's -- coming to the rescue, in terms of what we assumed, and I'll talk about what we assumed in a broader context.

But, in terms of what we actually incorporated into the MELCOR runs --

MEMBER REMPE: That's what I want to know.

MR. RECKLEY: Hossein Esmaili.

MR. ESMAILI: Well, I'm not going to talk about the MELCOR runs, but when I think this question

1 was asked about the penetrations, the sealant and
2 everything and I just have -- this is the experimental
3 programs as Bill was saying that was conducted in the
4 '80s, this was Sandia National Labs.
5 I don't know about that particular NUREG
6 that you're talking about, but you have two NUREGs
7 5096 and 4944 that deals with the seals and gaskets.
8 And, basically, they looked at both the,
9 you know, exposure to steam and air and in all the
10 analysis that they found out for EPDM seals, the seals
11 behaved, you know, in the temperature of 600, 650
12 Fahrenheit that that was the failure rate.
13 And, failure of the silicone seals was
14 about 500 degrees Fahrenheit in steam and about 700
15 degrees Fahrenheit in air.
16 So, the temperatures that we are seeing
17 here, we are limiting it to below 300 Fahrenheit from,
18 you know, the analysis that we are seeing.
19 MEMBER STETKAR: I'm not sure that's true.
20 When we see the Mark I and Mark IIs, they are
21 designing up to 500 and some odd degrees.
22 MR. ESMAILI: This is just the --
23 MEMBER STETKAR: These are strictly the
24 non-Mark I and Mark IIs? I often have times tried to
25 understand where I'm pigeonholed.

1 MR. ESMAILI: Unfortunately, I'm not an
2 expert in these seals, so I just asked, you know, a
3 review of these NUREGs that I just mentioned to you
4 and these are the temperature ranges. Anything above
5 500 degrees Fahrenheit, this is the minimum that they
6 expect failure for these silicone or other type of --
7 MEMBER POWERS: You cite a sensitivity to
8 atmosphere in the seal failures, air being a less
9 aggressive medium than steam. Do any of your studies
10 look at possible synergism between the ambient
11 atmosphere and the radiation dose? Hossein is
12 smiling.
13 MR. ESMAILI: Hossein is smiling?
14 I could hardly -- was that a question from
15 you?
16 MEMBER POWERS: You bet you.
17 MR. ESMAILI: Okay, I am not familiar. If
18 you want additional information, we can get that
19 information for you in the future. We just -- I'm
20 just responding to the earlier question on the
21 containment failure.
22 MEMBER REMPE: Well, let me be a bit
23 explicit on that earlier question. Information in
24 NUREG/CR-6906, Section 4.4.4 and 4.7.2 applies to
25 temperatures greater than 240C may cause degradation

1 and it gets worse if you go above 300C for containment
2 behavior.
3 And, what I'm asking is did you use the
4 old IPEs in the MELCOR analyses? Is that what the
5 containment strength was based upon or do you consider
6 degradation --
7 MR. ESMAILI: We did not consider it -- we
8 didn't consider that in MELCOR calculations. We did
9 not assume a thermal failure of penetrations in the
10 MELCOR calculation.
11 What we did is after the calculations were
12 run, we just looked at what the temperatures of the
13 structures were and showed that these temperatures
14 remained below 200 or 300 degrees Fahrenheit.
15 MEMBER REMPE: Okay. If you stayed below
16 200, I understand that.
17 MR. ESMAILI: Right. And the --
18 MEMBER REMPE: But, probably, you used the
19 old IPE curves is what I'm guessing. But, I didn't
20 know and that's what I was curious about.
21 MR. ESMAILI: It was not explicit enough.
22 MEMBER STETKAR: Hossein, I raised the
23 question in the Subcommittee meeting in terms of scope
24 of those assessments.
25 And, what I explicitly asked about was did

1 you look at containment penetrations in the way I
2 think of them including things like large ventilation
3 penetrations that have dampers or valves in them that
4 have internal seats?
5 And, I'm not talking about, as was said in
6 the Subcommittee meeting, things that are welded into
7 a steel liner or grouted into a concrete pipe
8 penetrations or electrical penetrations.
9 This would be the internal seats of
10 isolation valves --
11 MR. RECKLEY: In the aftermath of TMI,
12 there were a large number of NRC programs and they
13 were segmented.
14 And so, the containment penetrations
15 including mechanical bellows kind of seals and other
16 seals was largely done at Sandia and that's the body
17 of work that's captured and the bibliography of 6906
18 lists those.
19 When you get to the performance of a
20 valve, like the -- like a containment purge valve
21 inside the pipe, that work was done as part of another
22 part of the NRC program largely done at Idaho as part
23 of the qualification of mechanical and electrical
24 equipment.
25 And those test results are available and

1 they're documented in another set of program documents
2 related to, again, the qualification of equipment.
3 I did look through those. They show that
4 you can get leakage of those kind of valves. There
5 are factors to consider, containment isolation valves,
6 there's usually two, so you have to, you know, the
7 first one will take the brunt of the temperatures and
8 pressure.
9 And, the second finding related to the
10 containment purge valves was they held pretty good and
11 largely started to leak more as you started to cool
12 back down.
13 And so, we did look at that. I didn't
14 include that in the list of NUREGs we provided.
15 MEMBER STETKAR: And, please don't send
16 it.
17 MR. RECKLEY: Now, going back to Dr.
18 Powers' question, I'm trying to keep track of them
19 here, the program at Sandia, and I won't try to be an
20 expert and Dr. Powers probably knows much more about
21 it than I do, did include some effects of radiation on
22 the materials.
23 Those tests, as I understood it, were done
24 as many such tests are, they probably irradiated the
25 material and then tested them. And so, there wasn't

1 really, if you will, a test that showing both at the
2 same time pressures, temperatures, chemicals,
3 irradiation all in the same environment, but they were
4 done to include irradiation of some of the elastomers
5 before they were subject to the temperature and
6 pressure tests that were done. But, that's about the
7 extent of my knowledge on it, though.
8 MEMBER POWERS: To be sure, we know that,
9 for cable insulation and other polymer, that there is
10 a synergism between field and temperature.
11 We have some Japanese studies that say
12 there's synergism between temperature and steam and
13 radiation and whatnot.
14 So, it's not ipso facto follow that things
15 that survive a single environment will, in fact,
16 survive the convoluted environment.
17 We do know that as long as the seal is
18 compressed that it can undergo a very, very
19 substantial degradation and still remain a seal. The
20 problem always comes when things unseat and then try
21 to reseat in these degrading environments.
22 And, unfortunately, that's probably what
23 we had going on at Fukushima. And, one that we cannot
24 preclude occurring, it seems to me, in any accident
25 that where we're relying on sealing for a very long

1 time in the aftermath of the accident.

2 MR. RECKLEY: And, I think we'll have to

3 acknowledge that.

4 I will talk in a little bit, and again,

5 it's an important distinction between trying to model

6 and understand what's likely to actually happen.

7 And, incorporating that same thing into a

8 regulatory decision, because when I get to it in a

9 minute, I doubt any of what we're talking about would

10 change our outcome in terms of --

11 MEMBER POWERS: Yes, but I mean that's

12 exactly right. It's difficult at the state of

13 disknowledge to then immediately say and, yes, this

14 shows that there will be a significant safety

15 improvement.

16 That's a harder step to make when you're

17 grossly ignorant.

18 MR. RECKLEY: I've been called much worse.

19 MEMBER POWERS: I didn't call you grossly

20 ignorant, I called me grossly ignorant. I was

21 definitely employing myself, not you. You actually

22 can make those no significant improvement judgments,

23 I'm incapable.

24 MR. RECKLEY: So, what we did then, and

25 this is in the white paper, in the way the white paper

1 was laid out, it included a large amount of material
2 from the original SECY-15-0137, our initial
3 discussions. And then, we added some discussions and
4 then, as I'll talk later, and as Greg mentioned, we
5 added a few more enhancements and clarifications after
6 the Subcommittee meeting that we'll talk about.
7 But, this just summarizes basically, when
8 you look at the various containments, the shaded area
9 is simply showing that for Mark Is and Mark IIs the
10 containment performance aspects, other than hydrogen,
11 were addressed by the issuance of Order 13-109. And,
12 the Commission has decided that we can stop where we
13 are for Mark Is and IIs.
14 We then looked at the others which are the
15 BWR Mark III and the pressurized water reactorized
16 condenser and large dries just to see what was in
17 place for the various concerns that we identified when
18 we looked at the Mark Is and IIs being what's done
19 before core damage, what might happen after core
20 damage, whether the containment serves a collateral
21 purpose important to core cooling like it does for
22 Mark Is and IIs and whether there's other failure
23 mechanisms.
24 Again, the NTTF, the Near Term Task Force
25 recommendation was on venting which focuses on over

1 pressurization but we tried to expand our thinking a
2 little bit beyond that to say are there other failure
3 mechanisms that we should look at to make sure that a
4 substantial safety improvement is not kind of behind
5 the screen while we're focused on only the pressure.
6 And then, the hydrogen aspects both within
7 the primary containment and in other buildings.
8 MEMBER STETKAR: Before you leave this, I
9 found this a really useful part of the paper. I kept
10 coming back to this to kind of remind me how things
11 were partitioned.
12 One of the questions that I had is, in the
13 white paper, because that's all we've seen, you just
14 mentioned it also and it shows up on this matrix, it
15 says for ice condenser plants and, in particular,
16 because I'm more familiar with the large, dry -- PWR
17 large, dry containments, that the containment function
18 is not related to core cooling.
19 And, I guess I'm familiar with a number of
20 large, dry containments, as I said, I'm less familiar
21 with ice condensers, where so-called containment
22 accident pressure is needed to maintain core cooling
23 functions after LOCAs so that I have adequate net
24 positive suction head for systems that are taking
25 suction on the containment sump such that, if I lose

1 the containment sump function, I then lose my core
2 cooling because my pump's cavitate.
3 MR. RECKLEY: So, we have a specific
4 footnote to acknowledge that. But, in this particular
5 slide, we were -- I was thinking of more direct tie
6 like you have with RCCI and the suppression pool and
7 a BWR.
8 But we do have a footnote to acknowledge
9 that even large, dries might credit containment
10 accident pressure for the net cause of the suction
11 head of a pump.
12 MEMBER STETKAR: Good, and, yes, and I saw
13 that footnote in the white paper. I just wanted to
14 make sure that it's carried forward in whatever the
15 Commission paper is and we kind of got it on the
16 record here.
17 MR. RECKLEY: This is an insight from the
18 Mark I and Mark II work and shows in the bottom, if
19 you will, the results of the risk assessment and the
20 changes in risk that we see as we looked at various
21 alternatives related to three accident water addition
22 capabilities and on the far right, engineered filters
23 to release -- to reduce the release if you do vent
24 during a severe accident.
25 So, I wanted to -- this kind of goes to

1 the point we were talking about before, given the
2 frequency of severe accidents and other behavior that
3 we look at, it would take a fairly significant
4 shortfall to move you closer to where we would be
5 justified to take a regulatory action.
6 The quantitative health objective for
7 latent cancer fatalities $2x10^6$ is shown to kind of
8 give you a frame of reference.
9 I'll talk about it later, but we have
10 changed the paper to make it sound less like we're
11 comparing one to the other, but the QHO is used for as
12 a frame of reference as you're looking at the change
13 in risk.
14 But, when we're evaluating a potential
15 change to the regulatory requirements, we're really
16 looking at what the change is. And, you can see, in
17 this case, the changes that we looked at are
18 relatively small when looked at in the total risk on
19 the plant.
20 MEMBER POWERS: If I understand this
21 correctly, this shows you the change in risk
22 associated with a loss of containment function?
23 MR. RECKLEY: Well, in large part, it ends
24 up showing you that. What it shows is, for an
25 extended loss of power for a Mark I plant, what would

1 be the benefit, for example, of an alternative one is
2 the status quo which was Order 13-109 really, Order
3 12-50 basically, of adding capabilities for adding
4 water and then further to the -- on the graph of
5 engineered filters.
6 And, you can see that, and we've talked
7 about this when we talking about the development of
8 the guidance and then also the containment protection
9 and release reduction rulemaking.
10 MEMBER STETKAR: Before we get too far
11 here, just point of information that we have it on the
12 record, it's shown on this graphic, it's called
13 Alternative 1 is called status quo.
14 Most people think status quo means, I go
15 out to a BWR with a Mark I and II containment and it's
16 the way it is today. That's not status quo for this
17 particular exercise.
18 MR. RECKLEY: It's from a regulatory
19 perspective status quo.
20 MEMBER STETKAR: But, the regulatory
21 perspective required a hardened filter vent. So,
22 status quo means I have a hardened severe accident
23 capable -- I'm sorry, not filtered vent -- severe
24 accident capable vent that could withstand those
25 conditions inside the drywell. That is your notion of

status quo here.

MR. RECKLEY: That's right.

MEMBER STETKAR: Which is already a heck of a lot different than what people have today.

MR. RECKLEY: Yes. That is what people are doing now in order to comply with the order. But, yes.

MEMBER STETKAR: Well, people are invoking something to the right of what that is.

MR. RECKLEY: Yes.

MEMBER STETKAR: Because that could be achieved in principle if I had a hardened vent on the drywell that's capable of withstanding 1,000 degrees C, I'll use a really high value.

MEMBER POWERS: When I look at the releases to the site of iodine and cesium at the Fukushima site, and of course, all I have is the radiation data from out in the countryside, which some people have admirably reduced down into nice little time lines for me, when I look at that, I see a kind of a steady state of release of cesium, iodine and approximately equal release for actions and whatnot.

And then, it's punctuated by events that seem to correspond to depressurization events. And, when they occur in the relative magnitude of iodine

1 release exceeds that of cesium by roughly in order of
2 magnitude during a depressurization event, I quite
3 frankly don't understand that and I suspect your
4 MELCOR code doesn't understand that either. Is that
5 the case?
6 MR. ESMAILI: Can you repeat the question?
7 MEMBER POWERS: I'm sure I cannot repeat
8 the question.
9 It is, when we look at the Fukushima
10 releases as we infer them from the external radiation
11 monitor, we see that during the depressurization event
12 that the iodine release relative to the cesium release
13 skyrockets up during depressurization events, at least
14 the first few of them.
15 As you get later in time, it's actually
16 less as though you were getting some desorption
17 release due to iodine that was not capable for cesium.
18 My point in bringing it up is that,
19 depressurization events look like they are very
20 substantial events with respect to the hazard beyond
21 the site boundary because of the heightened iodine
22 release.
23 And, I indicated, to further the
24 perception that I'm grossly ignorant, that I don't
25 understand that and I speculated that, perhaps, MELCOR

1 doesn't understand that.

2 MR. ESMAILI: I guess if you don't

3 understand that, there's no point for me standing

4 here.

5 But, I can tell you --

6 MEMBER POWERS: Nice answer.

7 MR. ESMAILI: I can tell you what we did

8 in the MELCOR calculation, the MELCOR calculation we

9 did as part of the CPRR was a little bit different

10 than what happened at Fukushima because that was a

11 containment failure and depressurization was because

12 of that.

13 In the CPRR calculations that, you know,

14 we do many times on, we assumed that we are vented.

15 And, we are venting at the worst possible time.

16 And so, by the time things get into the

17 suppression pool, you have to open the vent and we are

18 getting, and you're right, you are getting much more

19 iodine out of the, you know, I mean, you have releases

20 on the order of, you know, 20 percent, even, you know,

21 without main steam line ruptures.

22 And because, you know, this is mostly with

23 the cesium, iodine and so, the cesium was about a

24 factor of ten lower, just because, you know, more of

25 the cesium is bound in the vessel and inside the

suppression pool.

But, my guess is that, you know, if you look at the SOARCA, for example, when they went without any venting, when they vented to containment failure and liner melt, et cetera, they were probably seeing less releases compared to what we are seeing because we are venting at the worst possible time.

So, a lot of things did not have time to settle or get scrubbed in the suppression.

MEMBER POWERS: Well, I mean, I bring it up because of his plot and questioning whether these deltas are true for things that lead to depressurization events relative to other kinds of events.

And, what you're telling me is, maybe not because just of the way you -- the conservative way you did the calculations.

MR. RECKLEY: That's what I heard.

MEMBER POWERS: That's what I heard.

MR. ESMAILI: These deltas are also obscured by the fact that in, you know, Marty is not here, but when Mary did his PRA, the fact is that, you know, maybe 40 or between 40 to 60 percent of the time, it didn't matter because we did not have Solve O or Solve M because they couldn't get the equipment

1 and the people didn't have it.

2 So, that's why it's not really affected by

3 -- even though it says, you know, water addition, but

4 the fact is that, you know, we don't have the

5 equipment, we don't have the people or --

6 MEMBER POWERS: That is an assumption?

7 MR. ESMAILI: Yes, so the PRA shows that.

8 MR. RECKLEY: So, again, after we gave

9 SECY-0137, 15-0137, the ACRS provided a letter report

10 in November the most noteworthy point number two was

11 a recommendation that we look a little more at

12 pathways for hydrogen migration to other buildings and

13 an expectation the NRC would continue its research

14 programs and international activities.

15 And, we did add specific language that all

16 of that international work and severe accident

17 activities and research we expect to continue. And,

18 we put that in the white paper.

19 We also added to the white paper more

20 recent work. There's an ongoing SOARCA analysis for

21 an ice condenser plant. So, we put in the preliminary

22 plots from that. That draft report is being prepared.

23 I assume the ACRS will have an opportunities to review

24 that work.

25 We asked research and they preformed a

limited analysis for a Mark III plant. And, we talked about that during the ACRS meeting.

And then, the one in the upper right is from the Surry SOARCA analysis just showing the pressure response fora large, dry containment.

We'll talk about hydrogen in a minute, but the real insight coming out of these more recent analysis was just a confirmation that the hydrogen igniters can prevent an early spike due to hydrogen combustion that could challenge the integrity of the smaller containments, the ice condensers and the Mark IIIs. And, I'll talk more about that in a minute.

MEMBER STETKAR: Bill, before we leave this, because we did have the opportunity in the Subcommittee meeting to discuss both the Mark III and the ice condenser plots that you show here and kind of the background behind those a little bit, the large, dry plot up in the upper right hand corner was part of your backup slides.

MR. RECKLEY: Right.

MEMBER STETKAR: But, we really didn't have time to discuss that. I went back, because I knew you were going to bring it, the reason it has the shape is that the SOARCA analyses assumed maybe too light but basically said, once you get up to a

1 pressure of precisely 2.17 times design pressure but,
2 I'll call it a little higher than twice design
3 pressure, you're going to open up enough of a leak
4 path such that pressure won't increase anymore and you
5 essentially relieve the pressure that way. That's why
6 that curve primarily has the shape that it has, right?
7 MR. RECKLEY: I'll look again.
8 MEMBER STETKAR: And that occurs at least
9 if you can't see the thing, for Surry, it occurs about
10 45 hours after the initiating event.
11 MR. ESMAILI: That is correct. That's the
12 type of damage that can happen in this type of
13 containment.
14 MEMBER STETKAR: Okay. And, I just wanted
15 to make sure that I understood --
16 MR. RECKLEY: And that's reflected why
17 it's not a sudden drop.
18 MEMBER STETKAR: Yes, yes, yes. Because
19 it's not a catastrophic failure, it's a tear or a leak
20 or they assumed that it was around the hatch, but they
21 just assumed that pathway. And, that's why you said
22 it's not a drop, but it's why it doesn't keep going up
23 from there on out either.
24 MR. RECKLEY: The equivalent of a drywell
25 hatch.

1 MEMBER STETKAR: I just wanted to make
2 sure that I understood that and the timing. It's
3 pretty clear that it's, if you look in the report, it
4 says 45 hours.
5 MR. RECKLEY: Right.
6 So, then we've briefly talked about this
7 before, the other thing we added to the white paper in
8 regards or in response to the ACRS observation was
9 this plot that we took out of the guidance developed
10 for Order 13-109 on the Mark Is and IIs.
11 And, again, the area in the green, the
12 containment penetration degradation area is the result
13 of the work done in Sandia in the '80s and shows the
14 range of conditions. I mean, it's not a bistable kind
15 of thing. So, it shows a range of conditions where
16 you might expect to start to see degradation and
17 failures of the containments.
18 And, this was considered as we did the
19 work for Mark Is and IIs, and then the work both by
20 EPRI or the industry and also the work that we did
21 that's reflected in the CPRR activities, showed that
22 the temperatures could remain below that 545 area.
23 And so, it was reasonable to assume that
24 the overall containment integrity would be maintained,
25 you wouldn't have massive leakage through the

structure or through the penetrations.

And so, within the paper, that meant when you did vent, the hydrogen would be going out the vent and not going into the reactor building. And so, that was the logic for Mark Is and IIs.

In the white paper, we took that discussion and this plot and, for the more recent work like the ice condenser, Mark III work, we also plotted the structural temperatures and, as I think Hossein mentioned earlier, we were seeing the temperatures were remaining below the 300 degree range.

And so, we think the containment integrity will be maintained sufficient that you won't have massive hydrogen leakage into the auxiliary building of a PWR or into the adjacent buildings for the ice condenser or Mark III.

MEMBER POWERS: When you do the Mark III calculation, what do you do about the bypass leakage from the drywell to the wetwell?

MR. ESMAILI: I have to check, Dr. Powers. I have to go back and see what we assume for the bypass leakage.

MEMBER POWERS: My recollection was that for -- in 1150, you found that to be a very substantial bypass leakage. Now, for, yes, for Grand

1 Gulf.

2 Presumably, your igniters keep up with

3 that with respect to hydrogen.

4 MR. ESMAILI: Yes, I remember. Yes, we

5 are -- we do have a bypass leakage directly from the

6 drywell to the wetwell.

7 As a matter of fact, this was, you know,

8 when we were looking at the GSI-189, this was one

9 source of hydrogen combustion that hot gases from the

10 drywell could go into the wetwell and ignite. You

11 know, that was a deterministic ignition source.

12 So, yes, we do model the bypass from the

13 -- the leakage from the drywell to the wetwell.

14 MR. RECKLEY: So, in the white paper, just

15 a summary of the white paper, we acknowledge we didn't

16 do the level of work that had been done for Mark Is

17 and IIs but we used the insights that came from that

18 work when we evaluated the other containments.

19 We looked at the body of work from the

20 '80s all the way up to the more recent SOARCA work.

21 We do -- that work did reflect the

22 benefits of the igniters in both Mark IIIs and ice

23 condensers. But, when we modeled the availability of

24 the igniters, it basically showed that they were

25 sufficient to resolve that issue or at least would

1 eliminate any additional regulatory action that we
2 would have to take to go above and beyond having the
3 igniters.
4 And so, we concluded in the white paper
5 that significant margins exist and that the changes in
6 risk that would come from any of the possible changes
7 to the venting or hydrogen control wouldn't be
8 warranted.
9 Based on the discussions --
10 MEMBER STETKAR: Bill, before we get -- we
11 can keep on this one for just a second.
12 It still remains, I think, I have to go
13 back and look at my notes, but I think for the Mark
14 III and the ice condenser plants, all of those or
15 those analyses were done on long term station
16 blackouts.
17 I have the additional time delay, if I
18 have to repower igniters, for example, or if I have to
19 hook up AC power to keep batteries charged, I have
20 that time delay.
21 I know in the Subcommittee meeting, we
22 brought up the notion of would the same conclusions
23 apply for short term station blackouts where I'm now
24 accelerated in terms of the progression of events
25 because I don't have the turbine driven core cooling

1 systems available.

2 That's not necessarily -- that doesn't

3 necessarily change the decisions about are igniters

4 good enough, but it certainly would change my strategy

5 in terms of how quickly I need to get those igniters

6 powered, especially if I have to repower them.

7 MR. RECKLEY: It does and it's a good

8 example how work done both by the industry and by the

9 NRC can carry back into the development of the actual

10 SAMGs.

11 And, the industry's going to talk later

12 about it a little bit, but there ware insights and

13 there were a fair amount of insights coming out of the

14 Mark I and Mark II work that went back into the

15 development of the SAMGs.

16 And so, you're exactly right, it may not

17 affect the regulatory, but it may very well actually

18 change what's done and what guidance is given to the

19 --

20 MEMBER STETKAR: Yes, implementations --

21 priorities in terms of timing and things like the

22 location even if it's, you know, do I park the diesel,

23 the spare diesel 25 miles away on a hilltop.

24 MR. RECKLEY: Right, right.

25 So, after the Subcommittee meeting there

1 were a couple observations and suggestions made and
2 so, I know we haven't given you the paper because it's
3 still being prepared, but we do have the draft
4 material so we know what we're going to change and I
5 wanted to highlight three things we're going to change
6 in this area as a result of the Subcommittee meeting.
7 One, we changed our language on the use of
8 the QHO just to stress, as Dr. Stetkar said, to that
9 it's an aggregate measure and that we're looking at
10 delta risks not a direct comparison, if you will, of
11 a single scenario against the QHO because that would
12 be inappropriate. It's always going to be some
13 fraction since any given scenario is only making up
14 some part of the total.
15 MEMBER STETKAR: Just my own personal --
16 it would be really good to get that green line off the
17 graphic that you put up here because, I don't care
18 what you say about it in words, people will look at
19 all of those little box and whisker plots and that
20 green line a couple of orders or magnitude higher and
21 draw the wrong conclusion. But, that's your paper.
22 MR. RECKLEY: Okay.
23 Well, and again, we have to go back to the
24 guidance that we're given in the NUREG which is
25 basically that's how we do consider it within the

safety goal screening.

So, but with the caveat that it is an aggregate measure and you've got to be careful and we just got it.

The other thing that we changed was we added some additional discussion largely out of the Surry SOARCA work related to the containment and hydrogen for large, dries.

MEMBER POWERS: What did that entail?

MR. RECKLEY: Well, the observation was made during the Subcommittee that the -- we really didn't change the white paper at all for large, dries from SECY-15-0137. And, the -- it may not be very satisfying, but my answer was, we didn't get any questions on large, dries, so we didn't change anything.

And so, it was pointed out that there was additional information and then the SOARCA study is a good example. So, we've added in some of that discussion of -- and we talked about it earlier, that you still do have time in a large, dry and so it is comparable to the other discussions we're having about why we wouldn't have a basis to take a regulatory action, require a hardened or a severe accident capable vent from a large, dry containment.

1 Then, lastly, what we added was some
2 discussion of short term station blackout. And, the
3 acknowledgment that the difference between a short
4 term and a long term is the short term assumes that
5 you actually -- that you also fail your turbine driven
6 cooling system.
7 And so, from the start of the transient,
8 you've lost everything whereas a long term station
9 blackout, you assume you have RCCI or the turbine
10 driven auxiliary feedwater for some period of time.
11 And so, it just speeds up the transient.
12 Of course, from a frequency point of view, it's a
13 little less frequency, less frequent because you are
14 assuming an additional failure.
15 I did want to make one note on this and
16 that is, for the work on Mark Is and IIs, which, by
17 the way, for the short term station blackouts, you can
18 fail containment as early as eight hours.
19 That was built into what we did for Order
20 13-109. And, when you look at the time line, the
21 guidance basically is telling the operators within
22 that time line that they need to do whatever they need
23 to do to make sure you can get severe accident water
24 addition and containment venting set up within the
25 eight hours before the reactor building would be hard

1 to access because the core will melt it through the
2 reactor vessel.
3 So, those are the three major changes that
4 we're adding. Again, it wasn't new work per se, it
5 was just bringing in references from other material to
6 beef up the discussion.
7 So, that's all I had on containment and
8 hydrogen. And, switch over to the next topic.
9 MR. SEBROWSKY: Thanks, Bill.
10 My name is Joe Sebrowsky, I'm a Project
11 Manager in the JLD. I work for Greg.
12 So, this topic is enhanced instrumentation
13 for reactor and containment for beyond design basis
14 conditions.
15 On this slide, this highlights what was in
16 the original SECY-15-0137 which included discussion on
17 the existing requirements for protection of
18 instrumentation.
19 We also added discussion in that SECy
20 about insights gained from proposed rulemaking for
21 mitigation of beyond design basis events.
22 One of the things that is discussed in the
23 SECY paper is the staff requirements memorandum
24 associated with the proposed rulemaking.
25 And, the staff requirements memorandum

1 told us three things when it comes to severe accident
2 management guidelines.
3 It directed the staff that SAMGs are not
4 a requirement, the proposed rule proposed SAMGs to
5 requirement. The Commission directed us otherwise.
6 It also directed in that SRM that SAMG
7 oversight should be placed in the reactor oversight
8 process. After the Subcommittee meeting that we had
9 on February 18th, we provided to Kathy or your staff
10 a February 23rd letter from our office director back
11 to NEI that outlined a process for how we would engage
12 industry to follow that Commission direction and
13 included near term and long term activities. I'll
14 talk about that in a little bit.
15 We mentioned in SECY-15-0137 that the
16 Commission gave us direction in that area and we are
17 now implementing that direction as described in that
18 letter that's been provided to the ACRS.
19 The last thing that the SRM said was or
20 directed the staff to do is ensure coordination
21 between the emergency operating procedures, the
22 extensive damage mitigation guidelines that come from
23 the (b)(5)(b) order, the FLEX support guidelines and
24 the severe accident management guidelines.
25 MEMBER STETKAR: Joe, just for the record

1 so we have it for everybody's awareness, it is
2 mentioned in the white paper that all -- every
3 licensee has submitted letters committing to the
4 institution of SAMGs, right?
5 MR. SEBROWSKY: That's correct.
6 MEMBER STETKAR: Okay.
7 MR. SEBROWSKY: And, that's documented --
8 MEMBER STETKAR: On the docket?
9 MR. SEBROWSKY: Yes. And, it's also with
10 specificity, if you look at the February 23rd letter
11 that we provided to the Committee, there's an
12 enclosure that gives you the accession number for all
13 the letters that we got from a hundred operating power
14 plants.
15 MEMBER STETKAR: I just wanted to make
16 sure we had that on the record for the purposes of the
17 Full Committee and this meeting.
18 VICE CHAIRMAN CORRADINI: Is there a
19 commitment to relook at them at some period to improve
20 upon what say we've learned from additional analysis?
21 MR. SEBROWSKY: What -- and, I was going
22 to talk about it later, but I'll talk about it now.
23 So, what you see in the letter is kind of
24 a two phased approach.
25 VICE CHAIRMAN CORRADINI: You can wait if

1 it fits in later.

2 MR. SEBROWSKY: It does, okay, might as

3 well bring it up now.

4 MEMBER STETKAR: Just for clarity, though,

5 Joe, when you talk about the letter, you're talking

6 about the staff's letter back to NEI with your two-

7 tiered approach --

8 MR. SEBROWSKY: Yes, that's correct.

9 MEMBER STETKAR: -- for regulatory

10 oversight? I think Mike may have been asking

11 something different about --

12 VICE CHAIRMAN CORRADINI: But, I don't

13 know what letter I'm asking about. What I'm going to

14 ask the industry, but I'm asking you is, when they

15 make a commitment to their SAMGs, which they already

16 have in some form and now we'll say more uniformity

17 and consistency, is there any commitment on their side

18 or expectation on your side that every three years,

19 every four years, every so often, they would relook

20 and things will be assessed based on additional

21 analyses? Because --

22 MR. SEBROWSKY: There's no explicit

23 provision, but to answer the question, one of the

24 things that we say in the longer term activity is

25 there, on the PWR side of the house, right now,

1 industry is developing PWR owners group guidelines
2 based on Fukushima lessons learned.
3 And, what you see licensees committing to
4 is they will implement their site specific SAMGs such
5 that it'll adopt those generic guidelines in the 2018
6 time frame.
7 MR. BOWMAN: So, I think your question's
8 more on a recurring basis after that.
9 MEMBER STETKAR: Steve?
10 MR. KRAFT: Yes, thank you, Mr. Chairman.
11 Steven Kraft from NEI.
12 Yes, all the units have filed the docketed
13 commitment as we've agreed to with the Commission.
14 Joe is exactly right about what the PWRs are up to.
15 The PWRs are a little scheduled out in
16 front of that. So, but there are a couple of things
17 here.
18 Clearly, it is in the units or the
19 utilities or the fleets best interest to adopt
20 guidelines as they come out. The owners groups have
21 their own training and their own way of doing this.
22 It happens regularly when something, you
23 know, something occurs. The Fukushima accident
24 obviously kicked off an initial set of changes and
25 there's a second set of changes.

1 The training is worldwide, it gets done.
2 But, the real question, I think, for the
3 people in this room is, what window does NRC have in
4 that to ensure that the commitment is being met.
5 And, I will tell you that what the
6 Commission directed the staff to do was to include it
7 in the reactor oversight process. There is an effort
8 being conducted right now through that whatever
9 organization in NRC does that with one of our task
10 forces. One of my colleagues in NEI is heading it up
11 and it is being in that direction.
12 And then, in the SAMGs, you get the TSGs
13 mentioned which there's a window for that.
14 So, I think that's all there. You're
15 getting the benefit of everything that would have been
16 done under a regulation without some of the baggage of
17 the regulation I think would have gotten in the way in
18 accomplishing this. It was always our view.
19 So, I think that's how that works.
20 And, I believe that there's a tremendous
21 amount of peer pressure in the industry, too, to be
22 able to do these things and to make sure they're done
23 correctly. And, we have here with us the chairman of
24 the committee that does this work in PWR owners group.
25 They take a great deal of pride and time and making

1 sure they get it right.

2 So, it's a long winded way of saying is

3 that Joe is right, we will -- those people will do

4 that and that's what that docket means.

5 VICE CHAIRMAN CORRADINI: Thank you.

6 MR. SEBROWSKY: Thank you, Steve.

7 So, to get back to the slide under staying

8 with the insights gained from the proposed rulemaking,

9 we talked about the direction that we received from

10 the Commission relative to the SAMGs.

11 One of the things to note that we placed

12 in the SECY for instrumentation discussion is the

13 proposed rule, although it recommended to the

14 Commission that SAMGs be a requirement, it did not

15 propose any instrumentation, enhancements to

16 instrumentation as part of that proposed rule.

17 And, it provided a rationale for that. It

18 essentially boiled down to four things.

19 SAMGs provide for use of alternate

20 instrumentation.

21 They include courses of action when no

22 reliable instruments are available.

23 They include consideration of instrument

24 uncertainties due to severe accident conditions.

25 And, they provide for use of computation

1 aids when direct diagnosis is not possible.
2 So, that was discussed in the SECY-15-
3 0137.
4 We also said in that SECY the ongoing work
5 that's happening right now, we'll talk about that in
6 a little bit and IEEE standard that's being updated
7 relative to instrumentation and what our plans are in
8 that area.
9 We talked about in that paper the
10 interactions with domestic and international
11 organizations.
12 We did provide quite a lengthy discussion
13 on the new reactor equipment survivability
14 requirements in accordance with Commission policy.
15 Fundamentally, new reactors have
16 expectations for enhanced instrumentation that differ
17 from operating reactors and we intend to continue that
18 based on policy decisions that were made by the
19 Commission in the '90s.
20 Lastly, consistent with what Bill said,
21 what the paper concluded is imposition of additional
22 requirements for enhancements to instrumentation are
23 not justified under the backfit rules.
24 This slide, I need to caveat, highly. We
25 added this slide from the ACRS Subcommittee meeting.

1 What this slide was meant to do is very limited. What
2 is was meant to do was to give the Full Committee an
3 idea of what instrumentation should be available to
4 plot and trend.
5 VICE CHAIRMAN CORRADINI: But, actually,
6 that wasn't a Subcommittee meeting. I interpreted
7 this as instrumentation that would not go out of range
8 when I entered in a beyond design basis event.
9 MR. SEBROWSKY: It's instrumentation that
10 we expect to maintain power. You won't have a
11 blackout condition, a blackboard like you had at
12 Fukushima.
13 MR. BOWMAN: So these are instruments that
14 the mitigating strategies order, which is a pre-core
15 damage order would ensure continuity -- would ensure
16 you have continuity power and to be able to use before
17 core damage takes place.
18 So, the message, you know, I don't want to
19 --
20 MR. SEBROWSKY: No, no, go ahead.
21 MR. BOWMAN: The message is that one of
22 the challenges they had a Fukushima was all the
23 instrumentation went away and there was just general
24 confusion even before core damage took place.
25 What the mitigating strategy order would

ensure through maintenance of these instrumentations is you'd be able to see a progression of the events up until the time core damage takes place.

And, Joe's going to talk a little more about what happens after that, but that's really the point of this slide and the instruments that are listed here.

MEMBER STETKAR: And, the environmental qualifications for this because it's all safety related.

VICE CHAIRMAN CORRADINI: Should they stay within?

MEMBER STETKAR: The environment will not exceed the environmental qualifications up until the time you get to core damage.

VICE CHAIRMAN CORRADINI: Okay, thank you.

MR. SEBROWSKY: Yes, that's correct.

And, when you see the set of instrumentation, you see a couple of things.

One of the things is it's very limited and it's basically telling the operators the minimum set of instrumentation that you should have available and the NEI guidance is very explicit, if you need to shed loads to keep -- maintain power to this instrumentation, you're expected to do that. If you

1 don't, then you keep as much instrumentation power as
2 possible.
3 But, it essentially tells you, for PWRs
4 and for BWRs, you either are or you are not removing
5 decay heat. That is the whole concept.
6 And, if you're not removing decay heat,
7 that you -- core damage is imminent and you need to
8 take appropriate action.
9 That -- the intention of this slide was to
10 show -- and we discussed this in the SECY paper, that
11 there were additional capabilities relative to
12 instrumentation that were added as a result of the
13 mitigating strategies order. It does not imply that
14 those severe accident capable.
15 If you go to slide 18, the ACRS letter
16 report comments on SECY-15-0137 suggested that we
17 should expand the discussion on calculational aids.
18 We should include a discussion of how severe accident
19 management guidelines and calculational aids are
20 capable of leading operators to take corrective action
21 even if minimal instrumentation is available or the
22 indicates are suspect.
23 And, provide a discussion on the
24 validation of what used to determine what
25 instrumentation is necessary before, during and

1 subsequent to a severe accident.

2 MEMBER STETKAR: And, just so I get on the

3 record here, the Fukushima Subcommittee did have a

4 half day briefing from the industry where we were

5 walked through an accident scenario as it developed up

6 through and including core damage and the short

7 extension post-core damage where we saw the transition

8 from EOPs to SAMGs, where we saw how some, albeit,

9 admittedly, limited number of calculational aids could

10 be used to correct for suspect instrumentation.

11 So, it was a closed session because it's

12 all proprietary information so I can't talk too much

13 about the details, but I just wanted to make sure that

14 we have it on the record that we did have some

15 exposure to that process.

16 MEMBER SKILLMAN: Joe, I'd like you to go

17 back a slide, please, to 18 -- to 17?

18 On this slide, what's interesting to me is

19 the -- on the BWR side, an operator can figure out

20 basically what's going on the reactor vessel or her

21 reactor vessel.

22 On the P side, get a pretty good idea of

23 what's available for your turbine driven P water pump.

24 And, you have some understanding of NPSA to your

25 coolant pump with 1 or 2 or 4 or 3 you're operating.

1 What you don't have is any indication of core level.
2 At TMI-2, we were using pressurizer level
3 as a surrogate to know that we recovered. Hence, when
4 we lost that level indicator, that's one of the
5 reasons we shutdown that fourth reactor coolant pump.
6 Perhaps with the BWR owners group -- PWR
7 owners group gentlemen here, or you, can answer, why
8 isn't there some action here, some instruments that is
9 a surrogate or reactor vessel water level?
10 MR. SEBROWSKY: So, I look to industry to
11 supplement anything I say. We'll go back to the
12 scenario for the mitigating strategies.
13 The mitigating strategy scenario is, it's
14 an extended loss of AC power and the loss of the
15 ultimate heat sink. So, you don't have reactor
16 coolant pumps.
17 One of the things that you don't see that
18 when you look at both the NRC's response, technical
19 manual and also the SAMGs, you don't see that
20 indication for a pressurizer level. You don't see red
21 lists reactor vessel level indication system.
22 This is a minimal set of instrumentation
23 that was thought needed to determine you are or you're
24 not removing decay heat.
25 You would not expect licensees to shed

1 loads unless it's absolutely, positively necessary,
2 loads such as power to red lists, a reactor vessel
3 indication system or the containment radiation
4 monitors that give you a supplemental indication on
5 whether or not you have core damage.
6 This is a minimal set to get you through
7 an extended loss of AC power and loss of ultimate heat
8 sink. It's to prevent a -- we're not sure if we are
9 or are not removing decay heat.
10 You would not want to go down to this
11 level of instrumentation in any situation if you could
12 avoid it.
13 MEMBER SKILLMAN: Okay, thank you.
14 MR. SEBROWSKY: This slide presents what
15 was in the white paper and the additions that we made
16 to the white paper to address stakeholder comments
17 that we received and also to attempt to address
18 comments that we saw in the ACRS letter report on the
19 SECY paper.
20 The first two are relatively simple. I
21 alluded to this earlier about the IEEE standard, 497.
22 That standard's being upgraded to provide additional
23 guidance on accident monitoring during degraded core
24 conditions.
25 That standard, we expect to be issued

1 sometime this spring. It was -- we had expected at
2 the time that we wrote the white paper that it was
3 going to be imminent, but it's been delayed a little
4 bit.
5 We do say in the paper, and we continue to
6 say, that after that standard has been promulgated,
7 our expectation is that we'll follow the reg guide
8 process and we intent up a Reg Guide 1.97 to reference
9 that new information in that standard.
10 The expectation is mainly that that will
11 be helpful to new reactors. But, if a licensee does
12 choose to voluntarily upgrade their instrumentation if
13 they're doing an instrumentation swap out and they
14 wanted to use that standard or reference that
15 standard, they could, but it would be voluntary.
16 We did expand the discussion of the SAMG
17 instrumentation to attempt to address the ACRS
18 comments and then, the input from the external
19 stakeholders.
20 The input from the external stakeholders
21 was, you know, was in two different areas. We had a
22 January public meeting where we invited comments on
23 the SECY paper. And, what we heard in that meeting
24 was a discussion that was more high level than the
25 discussion that the ACRS Subcommittee heard on January

1 the 18th.

2 But, it was a discussion of -- from

3 industry's perspective how they would implement SAMGs

4 and what they would do when instrumentation's either

5 suspect or unavailable.

6 We tried to capture that, and I'll talk

7 about that in the next couple of slides.

8 The other input that's germane is we did

9 receive from a member of the public a concern that in

10 the aftermath of Fukushima, we went out and did

11 inspections to look at SAMGs and how they were

12 implemented at a site by site basis. We did it for

13 all the operating power plants.

14 And, what we reported back to the

15 Commission, and it's documented in the SECY and the

16 proposed rule, is we saw a spectrum of capabilities

17 when it came to SAMG implementation from licensees and

18 operators fully understanding and knowing exactly

19 where to get the SAMGs and/or trained on it to

20 licensees that didn't quite meet our expectations.

21 And that was one of the reasons that we,

22 in the defense-in-depth had recommended the Commission

23 that the SAMGs be a requirement.

24 The comment from the member of the public

25 was, what are you doing to prevent the reoccurrence?

1 And, our answer to that is what the Commission
2 directed us to do and make sure that SAMGs are put in
3 the ROP and are inspected.
4 And, as I indicated, that's what we're
5 doing as documented in that February 23rd letter.
6 The next slide, and industry is going to
7 be talking about this in their presentation, there is
8 a tech support guidance document that goes with the
9 SAMGs.
10 The tech support guidance has four things
11 in it for interrelated documents, a control parameter,
12 assessment guideline, a plant status assessment
13 guideline, function status assessment guideline and an
14 emergency procedure guideline and severe accident
15 management guideline -- assessment guideline.
16 Within those four documents that are part
17 of the technical support guidelines, the one that is
18 most germane to the ACRS concern on enhanced
19 instrumentation is the control parameter assessment
20 guideline.
21 It has four subparts to it, validate
22 control parameter readings, monitor and trend the
23 important parameters, adapt frequency of monitoring
24 based on the rate of change of the monitored parameter
25 and perform guidance calculations to support the

1 parameter assessment.

2 If you go to slide 21, this is a subset of

3 the control parameter assessment guideline and, what

4 it would tell you to do in response to validating

5 instrumentation and these four bullets talk about

6 that.

7 This is, by the way, very consistent with

8 the guidelines that we, the NRC, use if an event like

9 this were to happen in the operations.

10 The use of other indications to confirm

11 readings, we have cautions in our manuals just like

12 the SAMGs that you never rely on just one source of

13 information, you try to use two or three sources to

14 ensure that your assessment of the core and

15 containment is accurate.

16 When actions are taken, parameters are

17 monitored to ensure instruments respond as would be

18 expected. Trends and changes in trends may be true,

19 even one absolute readings or values of readings are

20 not.

21 There was a discussion in what industry

22 walk through is. You may end up having problems with

23 certain level indications because they are not

24 qualified from the environment in which they're seen,

25 steam in the containment, for example.

1 There are corrections that are expected to
2 be made but, nevertheless, even though the absolute
3 value is off, if you're looking at a reactor vessel
4 level indication for a BWR, you would expect it to
5 increase if you believe you're pumping water into the
6 reactor.
7 And, the last one is an assessment to
8 consider the physical explanations for the instrument
9 readings.
10 If you go to this slide, what we thought
11 we heard at the Subcommittee meeting on January is
12 that we needed to supplement the technical support
13 guidance with the focus on computational aids.
14 So, what we tried to do is recognize that
15 the SAMGs are proprietary in nature. We couldn't
16 reference what computational aids are within the
17 SAMGs, but we could reference publically available
18 documents.
19 And, what we referenced is our response
20 technical manual, the NRC's response technical manual
21 that includes computational aids that are similar in
22 nature to that which you find in the SAMGs.
23 If you get back to some of the discussions
24 that industry had with the Subcommittee is based on
25 core conditions and when you see, you get to the top

1 of active fuel, we have computational aids that say,
2 this is how much water at T-0 you need to start
3 pumping into the steam generators or into a PWR into
4 the reactor vessel.
5 And, they are gross computational aids.
6 And, if you're not doing that, there's what the NRC
7 does as an independent tool is an assessment of the
8 fission product barriers. You either are -- we either
9 do or do not have confidence that you're removing
10 decay heat.
11 So, there is a set of computational aids
12 that are in that response technical manual that help
13 the staff do an assessment. And, in some cases, it's
14 based on limited or no instrumentation. It gets back
15 to a stopwatch.
16 What we tried to do to supplement the
17 discussion that we had in the white paper is to
18 reference that discussion. And, we -- when you go to
19 the last bullet, it gets to the SAMGs and supporting
20 instrumentation and calculational aids serve the high
21 level goal of restoring or maintaining cooling of core
22 debris and containment integrity.
23 And, the example that we use is, if you
24 assume you have minimal or no instrumentation and
25 containment and but you know the accident progression,

1 what our guidance would suggest you do is the same
2 thing that industry guidance suggest that you do and
3 that is pump water and try to either restore core
4 cooling or cool the debris.
5 And, the sequence of events is very
6 important and the timing is very important. So, we
7 tried to add some more discussion to be reflective of
8 what we heard during the January meeting.
9 And, that's all I have.
10 MEMBER STETKAR: Thanks, Joe, okay, Bill.
11 Any members have any other questions for
12 the staff? If not, we'll switch gears and NEI has a
13 presentation, in particular, the PWR owners group.
14 So, we'll ask them to come up and give us their
15 presentation.
16 MEMBER STETKAR: Randy, it sounds like
17 you're probably open. Could you just say hello? It
18 only sounded like you're -- Randy Bunt, anybody?
19 Randy Bunt?
20 MR. BUNT: Oh, I'm sorry.
21 MEMBER STETKAR: Okay, just to help us --
22 just say a few words like a couple of sentences or so
23 so we can make sure that we're picking you up fine and
24 that whatever you're speaking into is adjusted right.
25 MR. BUNT: (unintelligible).

1 MEMBER STETKAR: Yes, it's not -- Randy,
2 whatever you're doing, don't do more or something
3 because we're not picking you up?
4 MR. BUNT: (unintelligible).
5 MEMBER STETKAR: That's not good either.
6 I don't know what you're using for a microphone, but
7 you're breaking up. Mostly not good.
8 MR. KRAFT: It was definitely a cell
9 phone. Randy's available to answer questions.
10 MEMBER STETKAR: Oh, it's only for
11 questions? Okay. Then we'll try to muddle on as best
12 we can.
13 Steve, I'll turn it over to you.
14 MR. KRAFT: Well, thank you, John.
15 I'm Steven Kraft from the Nuclear Energy
16 Institute. Randy, when you're not speaking, if you
17 could mute your phone, please. There's some
18 background noise.
19 It is always a pleasure to come here and
20 you laugh, John, but it truly is. People at my office
21 wonder about me when I say those things.
22 MEMBER STETKAR: That's why I'm laughing,
23 Steve, it's --
24 MR. KRAFT: You know, there's a lot of
25 group self-hate in this world and there's too much.

1 So, in any event, we're pleased to come
2 back, appreciate the invitation.
3 I'm joined today by a well known figure,
4 Jeff Gabor, who is now properly from Jensen Hughes.
5 I'm told every time you bait him into
6 saying errant engineering, it costs him a buck, so,
7 you know, as often as you want is okay by me.
8 And, we have a new friend with us today,
9 Reed Labarge, from Westinghouse Electric representing
10 the PWR owners group. Reed will attempt to, when he
11 does his presentation, as to who is and his
12 background.
13 So, when we contemplated today's
14 presentation, we thought, well, we did in front of you
15 on this end and related issues at my lasts count about
16 4,500 times and you must be getting as bored with us
17 as could be.
18 So, we thought, well, we didn't need any
19 slides because we're not quite sure what it is we were
20 going to show you that we hadn't already done so.
21 And, on the specifics of these Group 2
22 issues that are remaining from the Tier 2, Tier 3
23 issues, we provided the staff comments on January 7th.
24 They have absorbed them and included relevant
25 information in the white paper.

1 There are apparently some changes they're
2 making to the white paper. We think they're right on.
3 I think we would just confuse the issue if we
4 attempted to explain the same things in our own words.
5 So, we're not going to try.
6 But, we thought that relative to the
7 instrumentation question that was discussed a few
8 minutes ago that the PWR group has not had the
9 opportunity to explain their process to you and that's
10 why we invited Reed to come along and do that.
11 So, let me turn it over to Reed.
12 MR. LABARGE: Thank you very much.
13 And, I'd also like to echo, thank you for
14 the opportunity for letting the PWRG kind of give you
15 a little sample of what we offer in terms of severe
16 accident management guidance with respect to
17 instrumentation.
18 So, again, I'm Reed Labarge from
19 Westinghouse Electric Company. I'm the technical lead
20 of the development of the PWRG SAMGs. And, I'd like
21 to just kind of give you a summary of what we have
22 with respect to instrumentation as it responds to a
23 severe accident.
24 So, before I get into the specific
25 guidance that we have with respect to instrumentation,

1 I thought it would be worthwhile to kind of show where
2 the technical support guide on instrumentation fits in
3 with the SAMG.
4 It's fully integrated. We have severe
5 accident guides available in the main control room as
6 well as in the technical support center.
7 The instrumentation guide is directly
8 referenced from both of those as well as from our
9 diagnostic process guide which we have a proprietary
10 handout version of if anyone would like to see, but we
11 would have to collect that at the end.
12 So, just as a summary of the SAMG
13 overview, we have the initial response happens in the
14 main control room. That's with SAG-1. When the TSC
15 is ready to take control, they progress to the
16 diagnostic process guideline which helps prioritize
17 the response to the severe accident based on plan
18 parameters.
19 So, the diagnostic process guide, given
20 that there are some known biases and other
21 uncertainties associated with instrumentation,
22 directly references the technical support guide on
23 instrumentation for additional insights.
24 In addition, strategies that are directed
25 from the main control room, SAG, as well as from

technical support center SAGs, are dependent on different plant conditions which also rely on some level of instrumentation.

So, those also have direct links to Technical Support Guide 1, which is what we'll be talking about for the rest of the presentation.

CHAIRMAN BLEY: Reed?

MR. LABARGE: Yes?

CHAIRMAN BLEY: I just want to interrupt you with an informational question for me.

These are slides for us?

MR. LABARGE: Correct.

CHAIRMAN BLEY: Are the new SAMGs or SAGs going to be flow chart form?

MR. LABARGE: They are not. Again, we have a copy, my associate in the back has a handout of what the diagnostic process guide looks like. And, it's really a matrix of parameters that helps prioritize what the response should be.

CHAIRMAN BLEY: Okay.

MR. LABARGE: What's the highest priority strategy. And then, the SAGs themselves are, they're actually the TSC SAGs are in a two-page format with guidelines on the right hand page.

CHAIRMAN BLEY: Okay.

1 MR. LABARGE: The left hand page has --
2 CHAIRMAN BLEY: Maybe after the session,
3 if we can get a chance to look at those, some of us
4 would like that, yes.
5 MEMBER SKILLMAN: A re-clarification,
6 please?
7 The way I read the left set of marks
8 there, the actions do not proceed until the TSC is
9 ready. Would you explain that, please?
10 MR. LABARGE: Sure.
11 So, the purpose of the TSC ready decision
12 box on this flow chart is to show that the main
13 control room retains command and control until the
14 technical support center is ready to take over.
15 So, SAG-1 actually does have severe
16 accident strategy guidelines in it. So, it gives the
17 main control room something to respond to the severe
18 accident before the TSC is ready to take over.
19 MEMBER SKILLMAN: Okay, thank you, Reed.
20 I understand.
21 MR. LABARGE: Okay. So, getting into what
22 our TSG-1 on instrumentation provides, TSG-1 provides
23 additional guidance for identifying alternate means of
24 measuring, inferring or confirming a plant parameter.
25 So, we're going to get into some of the

1 details here, but there are a few steps at the
2 beginning of the TSG.
3 The first is to review the available
4 parameter history. So, presumably, you've gotten to
5 this technical support guide either due to failed
6 instruments, unavailable indications, erratic
7 indications.
8 So, the step one is to make sure that you
9 have all available information and trends and, you
10 know, if things started getting erratic, well, what
11 did they look like before they were getting erratic?
12 Step two is a directional step. If
13 indication is actually available or not.
14 VICE CHAIRMAN CORRADINI: Can we just make
15 sure I understand what's going on?
16 So, in step one, the plant -- so, I don't
17 -- I'm not exactly sure how you enter into this. I
18 enter into this when there is some sort of abnormal
19 event and it proceeds to some level where this would
20 then be required, the plant staff would pick this up
21 and start going through the steps?
22 MR. LABARGE: Sure, let me explain.
23 So, this is a document that's only
24 available in the severe accident management
25 guidelines.

1 VICE CHAIRMAN CORRADINI: So, you enter
2 into it at that time?
3 MR. LABARGE: So, once you enter a severe
4 accident -- once you enter the SAMGs, you would be
5 directed to TSG-1 any time there was a concern about
6 instrumentation.
7 VICE CHAIRMAN CORRADINI: Okay, thank you.
8 CHAIRMAN BLEY: And, before you go ahead,
9 the thing is being passed around to us.
10 MR. LABARGE: Yes.
11 CHAIRMAN BLEY: Would it be inappropriate
12 to ask you a question about this?
13 MR. LABARGE: Yes.
14 CHAIRMAN BLEY: Yes, okay, we'll do it
15 later.
16 MR. LABARGE: Okay.
17 MEMBER REMPE: But, on your very first
18 one, where it says step one review available parameter
19 history, are all of the parameters that was shown by
20 the staff earlier, are they available to provide a
21 history immediately to the operators? Because some of
22 the parameters in the older plants, at least, are not
23 that way. And so, has someone thought through that
24 and that's way at this time?
25 MR. LABARGE: Reviewing the available

1 parameter history as part of the diagnostic process
2 guide, you'll be recording any available indication
3 and saving that information for trending.
4 So, whether or not it was available, you
5 know, it may be digitally previously, it will be once
6 you're in SAMGs, you'll be recording on a paper copy
7 what available, you know, indications that you have.
8 So, going back to the presentation, if
9 according to step two, if indication is not available,
10 so if an -- if a parameter maybe is reading blank, we
11 do direct to the appropriate FLEX support guidelines
12 for either restoring DC power or other power to
13 instrumentation or also for the FLEX support guide for
14 obtaining alternate means of reading a plant parameter
15 without power.
16 So, there is a direct link here and an
17 integration between the FLEX strategies and SAMG.
18 Additionally, in step 2(b)(2), we give
19 just -- this is more like a note of the most useful
20 tables of information that we'll be going through
21 later on within the technical support guide.
22 And then, you can proceed at that point,
23 you can proceed to the appropriate attachment for the
24 parameter that you came to TSG-1 for.
25 Go ahead.

1 So, this is kind of a summary of what type
2 of content is contained in TSG-1 for the PWR SAMGs.
3 And, I think you'll see that a lot of the information
4 here has already been indicated to you by the staff
5 and also by the BWR OG in their instrumentation
6 guidelines as well.
7 If you're -- if TSG-1 was entered to
8 confirm a plant parameter, we have pre-calculated
9 correction factors if that's appropriate for the given
10 parameter. Those are also calculated on a plant
11 specific basis.
12 We have alternate confirmatory indications
13 listed, expected response of a piece of
14 instrumentation based on other events or parameters.
15 Instrumentation calibration details and
16 measurement ranges, also severe accident
17 instrumentation reliability information.
18 If TSG-1 was entered because you didn't
19 have indication or if you'd like an alternate method
20 to measure and infer a plant parameter, we, again,
21 just to reiterate, we do have a link to the
22 appropriate FLEX support guidelines for signal
23 restoration and alternate measurement methods.
24 Going back a little bit to a comment that
25 was made earlier regarding vessel water level

1 measurement which wasn't listed as one of the FLEX
2 support guide parameters, that would be an example of
3 a parameter that we do have guidance for within the
4 severe accident management guidelines.
5 So, if there's alternate means of locally
6 measuring or restoring a parameter that wasn't
7 included in FLEX, we also list that here in the tech
8 support guide. And, you'll see that also on the next
9 slide.
10 So, there are 13 parameters that we have
11 in the generic guidelines. We also provide guidance
12 to the plants that, if there are additional parameters
13 that they feel are important for their plant, those
14 can be added during plant specific implementation.
15 And, as you can see here, Attachment D
16 contains reactor vessel water level indication.
17 Some of the sample information that we're
18 going to be providing on the next few slides is
19 actually with respect to steam generator water level,
20 but a lot of that will be similar to some of the other
21 parameters.
22 VICE CHAIRMAN CORRADINI: And, just so I
23 get this, these attachments are where you go if the
24 answer is no, that something seems awry?
25 MR. LABARGE: You actually enter this --

if you've entered TSG-1 for a particular parameter, whether or not you have indication available or not, you would still proceed to the appropriate attachment.

So, as I mentioned, this was a relatively brief presentation, so the next few slides are just some sample information.

This sample is actually for steam generator level and I realize that some these might be a little difficult to read on the screen, but this was just to show that we have a table for measurement methods. So, we have a primary and a secondary, as well as a list of alternate indications.

You know, whether that's a FLEX strategy to measure something at an instrumentation rack or alternate parameters that might not be immediately obvious that you could use to help infer some sort of -- to give you some sort of a feeling for what your plant condition is.

The second table on this slide is the impact of the accident on instrumentation. So, this is where we get into some of the reliability of different pieces of instrumentation to a severe accident.

So, some pieces of instrumentation may not be expected to survive severe accident environments

1 and that type of information would be included here.
2 This is our -- this is from, again, from
3 the generic SAMGs and this is our -- these are our
4 tables of correction factors.
5 We actually have a background document
6 that gives guidance on how to calculate these on a
7 plant specific basis. But, it will show you if you're
8 getting a specific indicated value, how you should
9 adjust that value based on different containment
10 conditions in this case.
11 So, this -- for a steam generator water
12 level, it's assumed that there's a -- it's calculated
13 using a delta P cell inside containment where, when
14 the containment heats up, you get -- you would get a
15 known instrumentation bias because of the reference
16 life.
17 So, you know, we give instruction to the
18 -- during plant specific implementation to pre-
19 calculate these values as they're available in your
20 severe accident management guidelines.
21 MEMBER BALLINGER: Is Attachment D
22 available as part of the writeup that we have here,
23 that we can look at? That's the reactor vessel water
24 level one?
25 MR. LABARGE: It is not. The sample

1 parameter -- we didn't know that there were going to
2 be questions about that, so we picked a representative
3 parameter and the parameter we chose for this
4 presentation is steam generator water level.
5 MEMBER BALLINGER: Can you provide it?
6 MEMBER STETKAR: Let's talk about -- I
7 have some other things to discuss with NEI and the PWR
8 folks after -- rather than take up the Full
9 Committee's time. We can wait until after the
10 meeting.
11 We'll eventually get to it, Ron, just not
12 here.
13 MEMBER REMPE: At a high level?
14 MR. KRAFT: Why does that not encourage
15 me?
16 MEMBER STETKAR: You said you really like
17 to come here.
18 MR. KRAFT: I do.
19 MEMBER SKILLMAN: It's a 10^{14} question
20 there.
21 MR. KRAFT: Yes, I know, well 15.
22 MEMBER SKILLMAN: Yes, 15.
23 MEMBER REMPE: At a high level, I mean
24 you've talked about plant specific implementation, can
25 you tell us where you are? I mean, have you gone to

1 any plants yet? How many have you gone to? And who
2 all's there, the owners group there? The plant staff
3 is obviously there. Is NRC there at all during the
4 plant specific implementation?
5 MR. LABARGE: Let me just give a quick
6 summary because the PWRG SAMGs that are represented in
7 this presentation were formally published last week.
8 Right, so Revision 0 was formally
9 published last Friday.
10 So, we did, as part of the development of
11 the generic SAMGs, we had SAMG validations performed,
12 generic SAMG validations performed at one of each of
13 the three PWR and Triple-S vendor types, so a
14 Westinghouse plant, a CE plant and a BMW plant.
15 As part of the validation, a simplified
16 version, we'll say, of the SAMGs, a partially plant
17 specific version of the SAMGs was developed to
18 facilitate our validation of the generic guidelines.
19 So, to answer your question, I think, you
20 know, no one has implemented plant specifically yet.
21 But, the industry is aware and these are now formally
22 published.
23 MEMBER REMPE: Thank you.
24 MR. MIRZIO: Can I add something real
25 fast? This is Damian Mirzio from the PWROG.

1 With respect to the request for Attachment
2 D, we'll look into releasing that to you guys.
3 MR. LABARGE: Thank you.
4 So, this is my final slide and, again,
5 this is just some additional sample information that
6 shows the -- for a steam generator water level, what
7 a typical range would be indicated and useful ranges
8 for the -- both for a steam generator wide range level
9 and narrow range level.
10 So, there's nothing in particular on this
11 slide that I wanted to discuss. I just thought it
12 would be useful to show an example of what we have in
13 our technical support guide.
14 CHAIRMAN BLEY: Do you mind going back to
15 the previous one?
16 MR. LABARGE: Sure.
17 CHAIRMAN BLEY: I want to ask a naive
18 question. These correction factors, we have them for
19 normal operating conditions, depressurized to 400
20 atmospheric pressure and then versus the orange and
21 red conditions, which, right now, I don't know what
22 those are. It's not a simple function of just
23 pressure and temperature, it's something more complex
24 than that?
25 MR. LABARGE: No, I would say it is

basically related to containment temperature and pressure.

CHAIRMAN BLEY: The reason I'm asking is because this gives me a big bunch of hunks to go and get something that's covering a range.

Back in the old days, if I had something like this, somebody would have drawn up a nice nomogram for me and I would have said temperature, pressure, zoom, there's my correction factor and I would have used it. It would have been known everywhere, it would have been more accurate.

I'm curious about why we've gone to these chunky things instead of something which, to me, but that's just me, would have been a whole lot simpler.

MR. LABARGE: Yes, we didn't develop, you know, the type of curve I think that you're talking about for these values.

This is in terms of steam generator water level, if you're at extremely high DPG orange condition and red condition are both above design value.

CHAIRMAN BLEY: Yes.

MR. LABARGE: So, these are the types of correction factors that we wouldn't normally expect operators or people in the TSC to be familiar with.

1 The type of resolution of an exact value,
2 the PWROG didn't feel that it was necessary to go into
3 that level of detail.
4 CHAIRMAN BLEY: Well, the reason I ask is,
5 and maybe it's the way people are trained, for me, a
6 nomogram with three lines on it, observe pressure,
7 observe temperature correction, would have been a
8 whole lot simpler than this thing where I've got to go
9 here and here and here. So, that's all I was asking
10 and that's enough, just let you think about it some
11 time because I think it's a very simple picture.
12 And, with a certain kind of plant used to
13 have those kind of nomograms sitting around.
14 MR. KRAFT: Well, you wouldn't --
15 CHAIRMAN BLEY: Of course, in engineering
16 school, you used to be taught how to make them.
17 MR. KRAFT: Because you wouldn't know how
18 complicated it is until see the document, but I think
19 what Reed was getting at is that you can be too
20 accurate and be mislead by the accuracy of that line.
21 I mean, there's a chart in Mark's handbook
22 that does exactly what you're talking about. It's not
23 a nomogram, it's just numbers.
24 But, again, I think what you're getting at
25 is you need bands of information because, otherwise,

1 you're -- oh, it's right here and you're going to
2 argue is it this much or that much? You know, you
3 want chunks so you know just put water in or don't put
4 water in. That's the way I kind of see it.
5 CHAIRMAN BLEY: Okay. I'm -- I'd look at
6 places where we've had, you're going to get two of
7 these lights and one of those and this pattern stuff
8 and here, we've got kind of pattern stuff, and it's an
9 error prone thing for people.
10 MEMBER STETKAR: Let me see if I can get
11 us back on track because we're delving off into
12 details of their SAMG structure, so thanks.
13 MR. LABARGE: Yes, I believe we've gone
14 through all the slides that we had for this
15 presentation.
16 MEMBER STETKAR: Steve, you have something
17 to wrap up?
18 MR. KRAFT: I'll just say thanks very much
19 and answer any questions you have. A few of us may be
20 here for a little while. I've got to get downtown and
21 --
22 MEMBER STETKAR: Anything more for Reed or
23 Steve? If not, what we'll do, the pops and crackles
24 notwithstanding that would lead you to believe the
25 public line is open, but it's not the public line,

1 what I'd like to do is get the public line open and
2 while we're doing that, ask if there's anyone in the
3 room who would like to make a comment? If so, please
4 identify yourself and do so.
5 Ed?
6 MR. FULLER: This is Ed Fuller from the
7 Office of Research.
8 On one of your slides, you showed the
9 prioritization of the various SAMGs from number three
10 onward. I know it wasn't a subject of this particular
11 presentation, but one of them on there was related to
12 hydrogen control. Does that apply to large, dries --
13 MEMBER STETKAR: These are comments from
14 the public. This is not a question and answer period.
15 MR. FULLER: Well, you didn't give the NRC
16 staff a chance to comment on this presentation, so I
17 thought I was a member of the public for that purpose.
18 Am I wrong?
19 MEMBER STETKAR: No, you may comment, you
20 may comment, but we're not going to entertain a
21 question and answer period. That's not what we do.
22 MR. FULLER: Oh, you just want a comment?
23 MEMBER STETKAR: Yes.
24 MR. FULLER: Okay. My comment is that it
25 doesn't make a whole lot of sense to me to apply that

1 to the large, dries in this country. And, if the
2 speaker wants to speak to me after this portion of the
3 meeting is over, I'll be happy to hear his answer.
4 MEMBER STETKAR: That's fine, thank you.
5 Anybody else like to make a comment, is
6 so, come up, identify yourself.
7 If not, what? Damian?
8 MR. MIRZIO: This is Damian Mirzio, again.
9 This is just a clarifying comment on the
10 reasons for the -- we call them quote, unquote chunks.
11 A lot of that has to do with the preventing analysis
12 paralysis too detailed of items that are included in
13 the SAMG during high stress situations. So, it's an
14 effort to reduce that analysis paralysis situation.
15 MEMBER STETKAR: Okay, thank you.
16 Anybody else in the room? If not, we'll
17 open it up to see if there are any comments from
18 anyone on the bridge line. I'm not hearing pops and
19 crackles, so I have to do this.
20 If there's anyone from the public on the
21 bridge line, could you do me a favor and just say
22 hello? That's the only way we have to verify that the
23 line is open.
24 That's always troublesome. Anyone on the
25 bridge line? Okay, well, I'll assume that it's open.

I've been told it's open. I'll assume there's no one out there.

With that, again, I'd like to thank the staff, thank the industry for the presentations and, Mr. Chairman, I'll turn it back to you.

CHAIRMAN BLEY: Thank you.

Three minutes to spare, you didn't need to cut me off.

MEMBER STETKAR: But, we're still on the record and, dang, it was fun.

CHAIRMAN BLEY: I'd only point out that you asked a very superfluous question at the end. Having no answer didn't affect anything.

I need to ask you before we recess, are you ready to go into read towards your letter?

MEMBER STETKAR: I am.

CHAIRMAN BLEY: Fifteen minutes or do you need some more time here?

MEMBER STETKAR: You know, I could start now. No, I don't need more time than that.

CHAIRMAN BLEY: Okay.

MEMBER STETKAR: I just want to --

CHAIRMAN BLEY: Okay. We will -- we're going off the record I think for the week, right? We don't have anything tomorrow to be on the record?

1 Okay, we will be going off the record for good and we
2 will have in house working sessions from now on.
3 We'll start on our letter writing on the letter
4 affecting this material that we just heard and at
5 3:15.
6 We'll recess at this time until 3:15.
7 (Whereupon, the above-entitled matter went
8 off the record at 2:58 p.m.)
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U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

ACRS Meeting:

Non-power Production or Utilization Facility (NPUF) License Renewal Rulemaking

March 3, 2016

NRC Staff Presenters

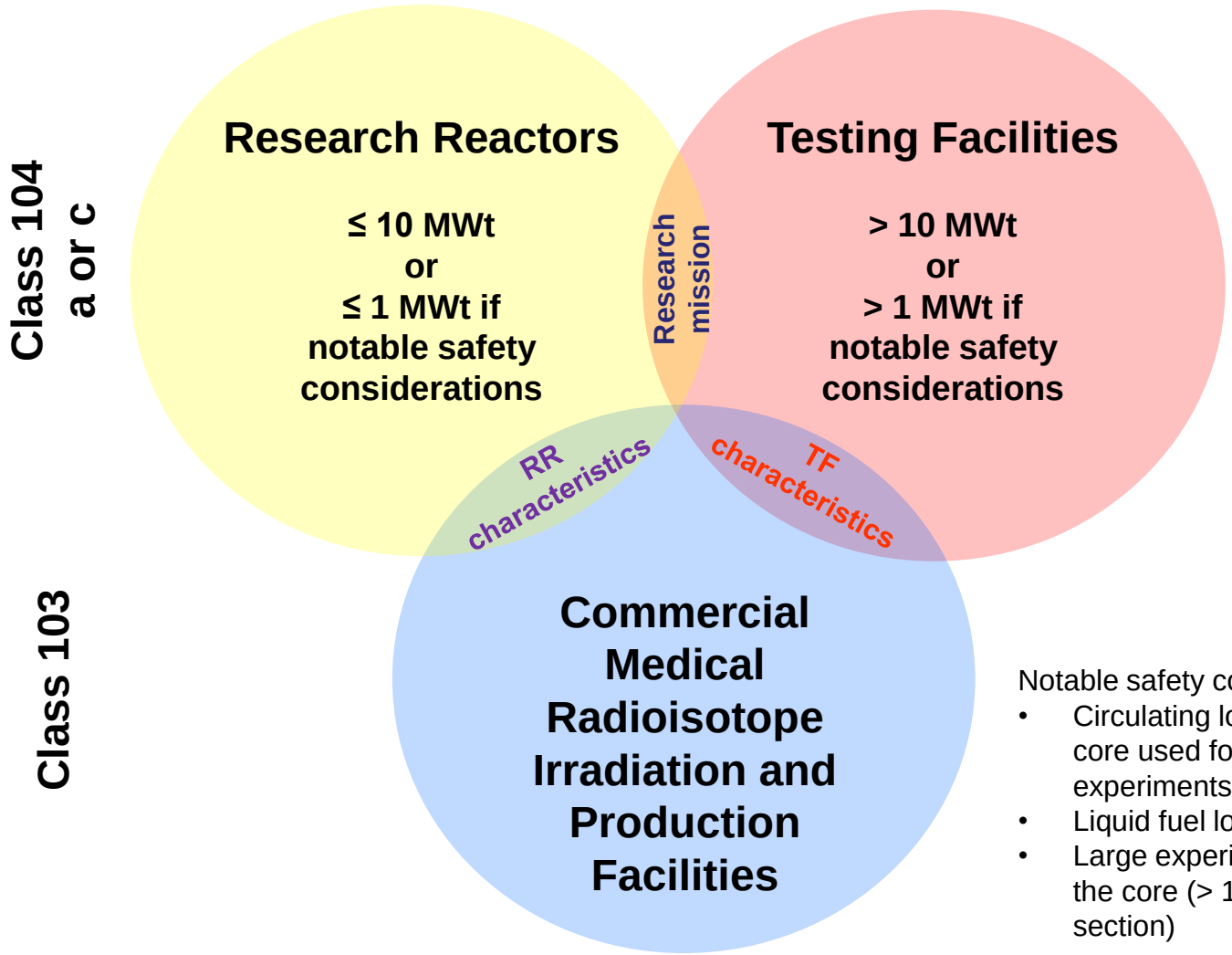
- Robert Beall, NRR: Rulemaking PM
- Duane Hardesty, NRR: Technical Lead
- Al Adams, NRR: Branch Chief - Research & Test Reactors Licensing

Purpose of the NPUF Proposed Rule

- Implement Commission direction to streamline the license renewal process by establishing a more efficient, effective and focused regulatory framework
- Fix existing short comings in regulations for non-power licensees

⇨ 9 rulemaking objectives

Relationship of NPUF Entities

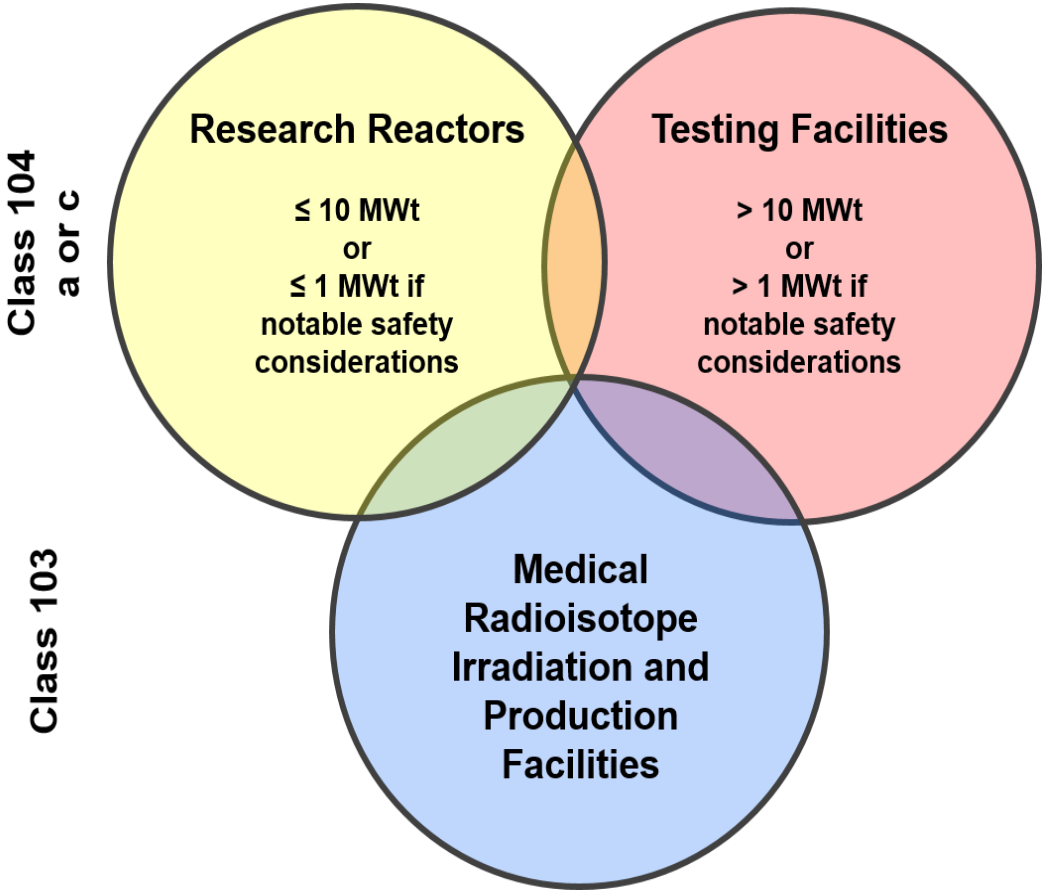


Notable safety considerations:

- Circulating loop through the core used for fuel experiments
- Liquid fuel loading
- Large experimental facility in the core (> 16 in^2 in cross-section)

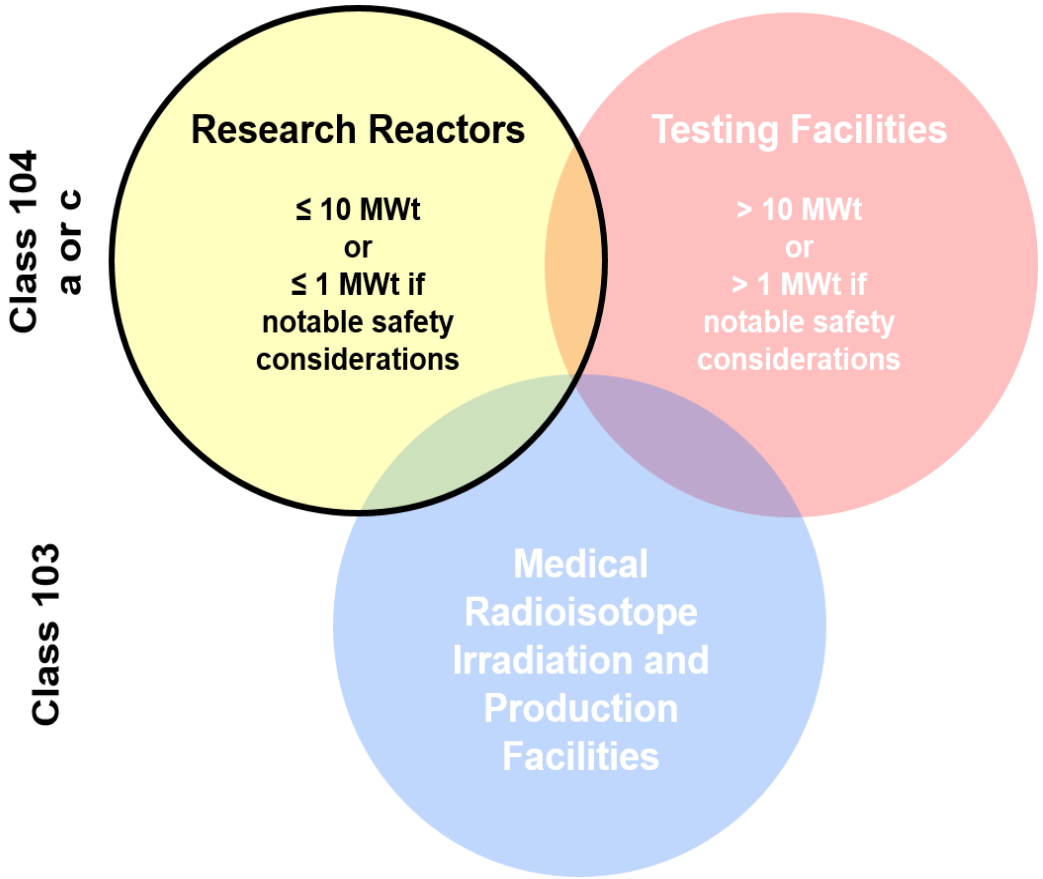
1. Create a Definition for NPUF

- Revise 10 CFR 50.2, Definitions, to establish a single term to capture all non-power facilities licensed under part 50
- Ensure clarity and consistency for the applicability of NPUF regulations



2. Eliminate License Terms

- Exempt Class 104a and 104c NPUFs, other than testing facilities, from 40-year fixed term in 10 CFR 50.51
- No license term specified in AEA for Class 104 NPUFs
- Consistent with AEA's minimum regulation standard
- Reduce burden for licensees and NRC, but maintains public health and safety



No Notable Safety Considerations

- Low power levels of 10 MWt or less
  - small fission product inventory
  - small radiological consequence for maximum hypothetical accident
- Low energy systems
  - low operating power and temperatures
  - minimal decay heat
- No significant aging considerations
  - simple designs
  - proactive aging management / aging-related surveillance requirements
  - loss of coolant is an analyzed condition
- Slowly evolving licensing basis
  - Very low number of design changes each year
  - Few rulemakings apply

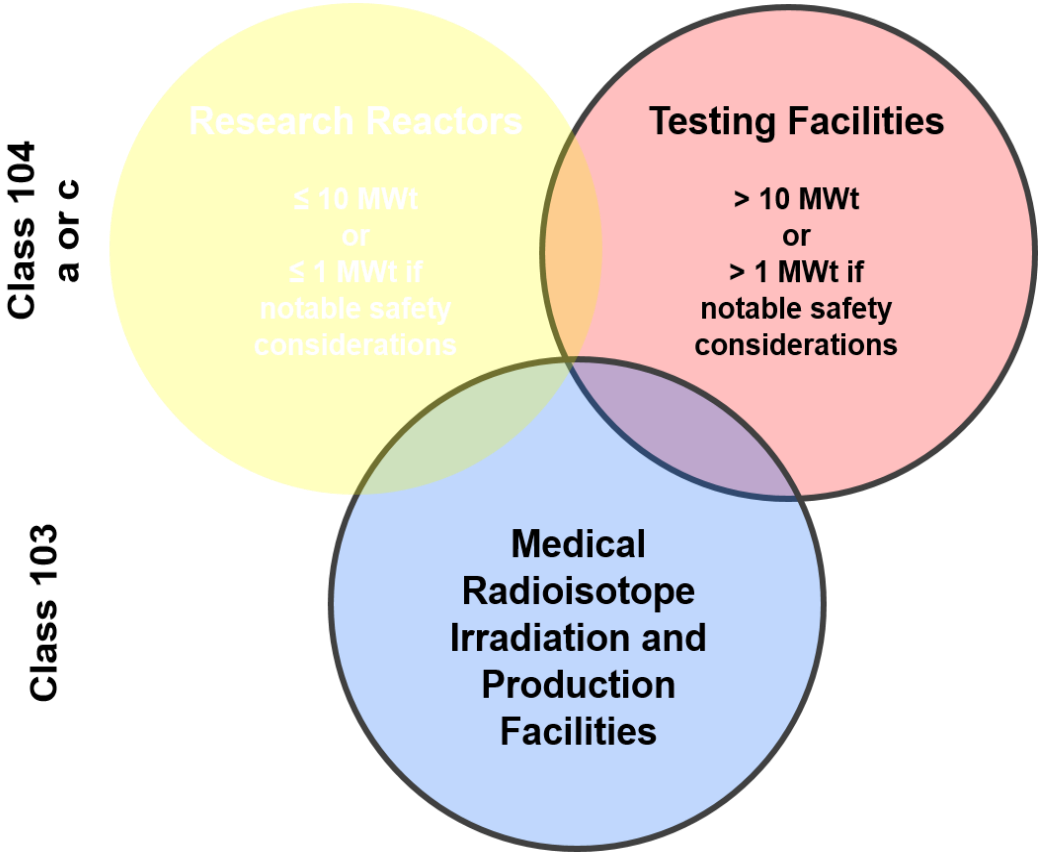
No Nexus between License Renewal and Safety

Class 104a or c, except testing facilities

- In light of the following measures:
  - NUREG-1537
    - License renewal under NUREG-1537
  - Inspection program
  - Technical specifications
  - Existing reporting requirements
    - Safety issues with SSCs
    - Maintenance activities
  - Proposed rule requirements

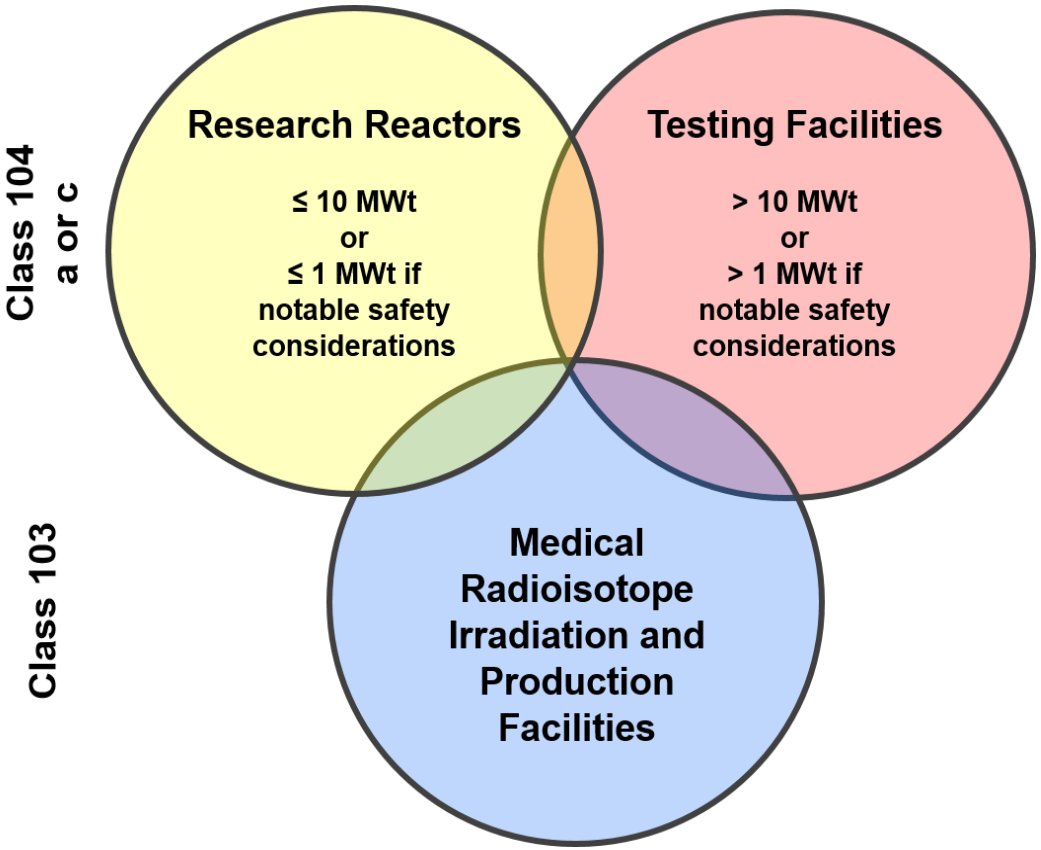
3. Define the License Renewal Process

- Consolidate license renewal requirements under 10 CFR 50.135 for testing facilities and NPUFs licensed under 10 CFR 50.22
- Clarify license renewal process
- Enhance regulatory efficiency



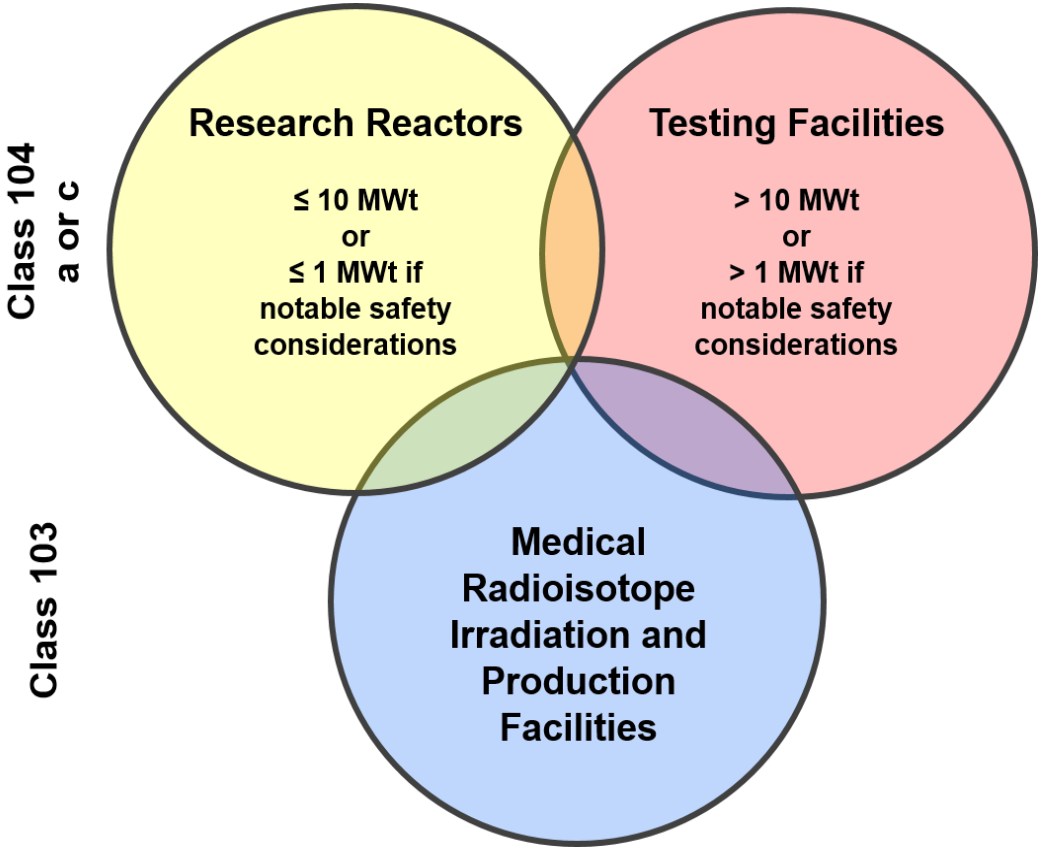
4. Require Updated FSAR Submittals

- Extend applicability of 10 CFR 50.71(e) to NPUFs
- Ensure timely documentation of changes to licensing basis
- Benefit knowledge management, NRC's inspection program, and licensee operator training and exams



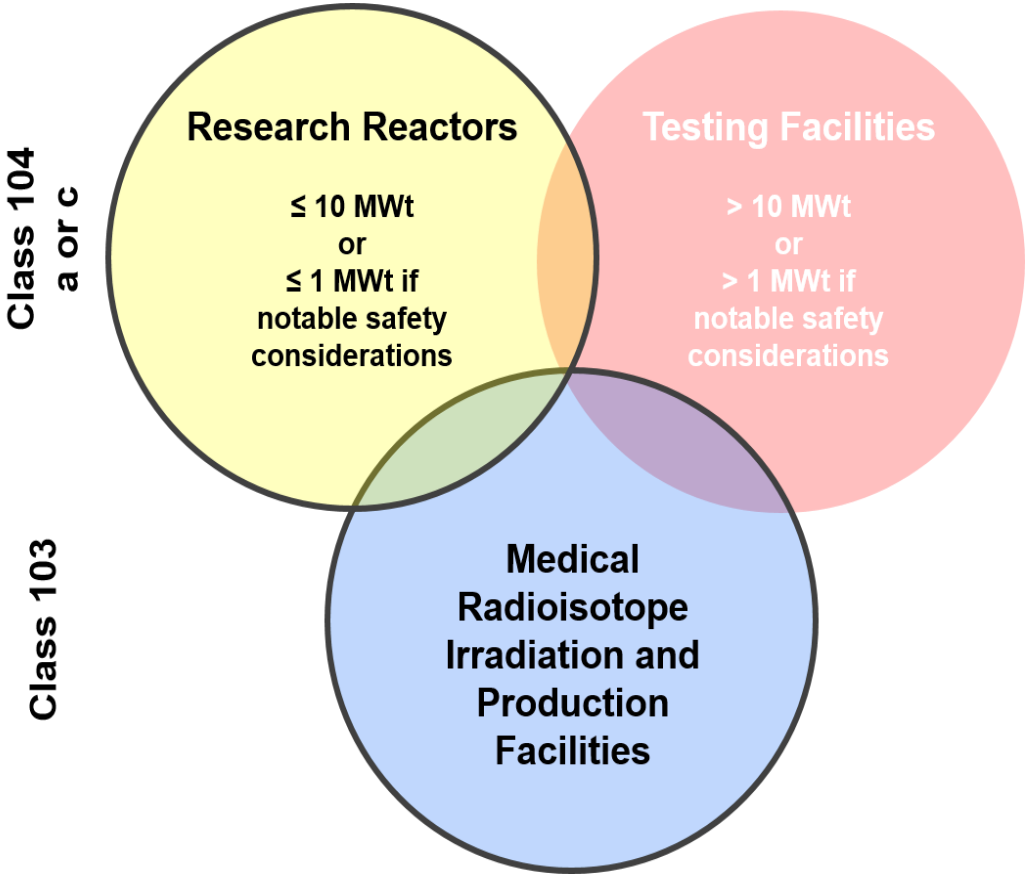
5. Amend Timely Renewal Provision

- Create two-year timely renewal for Class 103 and testing facilities and exempt Class 104a and 104c NPUFs, other than testing facilities
- 30 days in 10 CFR 2.109 is not a sufficient period of time for adequate assessment of license renewal application
- Two years would provide sufficient time and allow facility to operate under current license terms



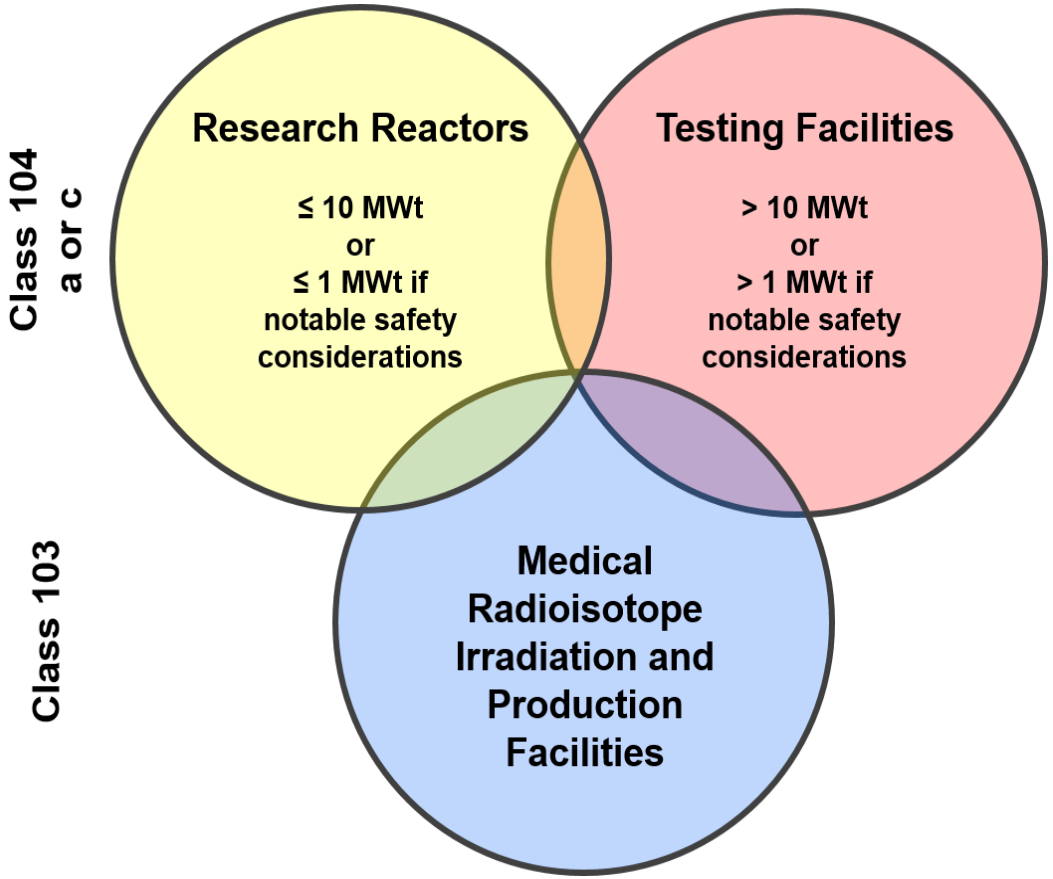
6. Provide an Accident Dose Criterion

- Create new accident dose criterion for NPUFs, other than testing facilities, in 10 CFR 50.34
- Part 20 public dose limits are unnecessarily restrictive as accident criteria
- Proposed criterion would align with early phase EPA PAG and provide adequate protection from unnecessary exposure to radiation



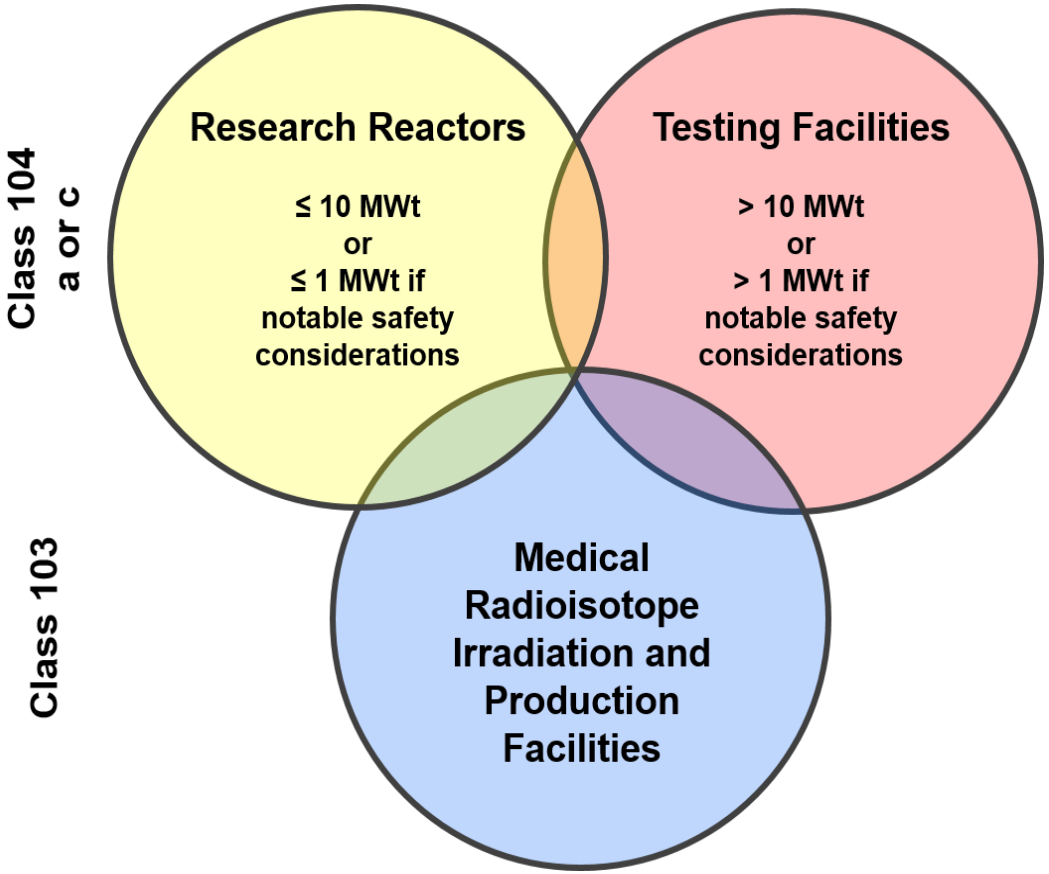
7. Extend Applicability of 10 CFR 50.59

- Extend applicability to NPUFs regardless of decommissioning status
- 10 CFR 50.59 currently is not applicable to NPUFs once fuel is moved offsite
- Avoid burden of issuing license amendments



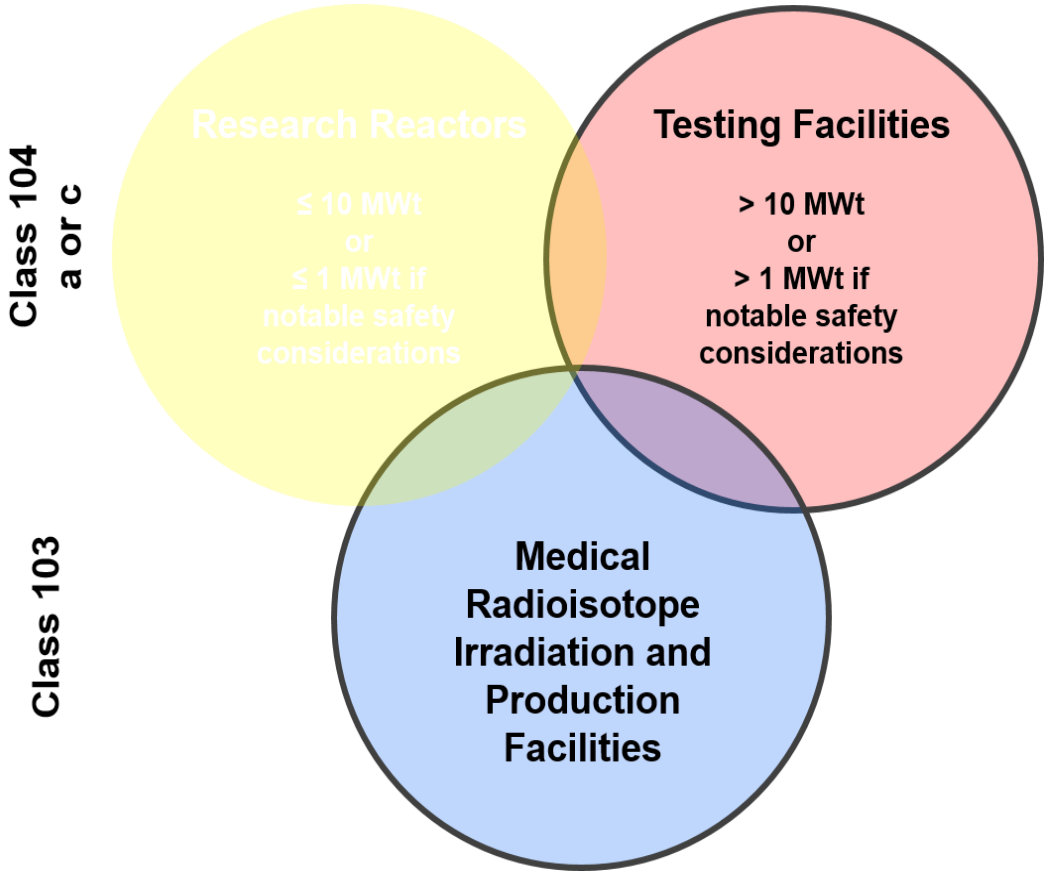
8. Clarify Existing Environmental Reporting Requirements

- Add requirement in 10 CFR 51.56 for NPUFs to provide an environmental report per 10 CFR 51.45
- Historically, NRC has relied on 10 CFR 51.41 to collect "environmental information"
- Improve consistency and clarify Part 51 requirements for licensing actions



9. Eliminate NPUF Financial Qualification Information Requirement

- Eliminate 10 CFR 50.33(f)(2) financial qualification requirement at license renewal only
- Primary means to ensure safety is through NRC's oversight and enforcement programs
- Reduce licensee burden without compromise to public health and safety



Summary of the NPUF Proposed Rule

- **Costs and Cost Savings (Undiscounted)**

	NRC	Licensee
Implementation Costs	$720,000	$140,000
Operations Costs	$1.8 million	$1.6 million
Cost Savings	$12 million	$5.5 million
Net Benefits (Cost Savings – Costs)	$9.4 million	$3.8 million

- **Total Net Benefit (Undiscounted): $13 million**
  - 3 Percent discounting: $8.9 million
  - 7 Percent discounting: $5.3 million
- **Backfit Considerations:**
  - Section 50.109 does not apply to NPUFs
  - Section 50.109 not applied to this proposed rule

NPUF Proposed Rule Package

- SECY Paper
- *Federal Register* Notice
- Regulatory analysis & backfit discussion
- Environmental assessment
- Draft Reg Guide DG-2006
- Rulemaking package currently in review by OGC
- Additional supporting NPUF rulemaking documents
  - Congressional letters
  - OMB statement
- NRR/DPR supporting NPUF documents
  - Project Manager Handbook Sections 4 & 5 updates

NPUF Rulemaking Schedule

- Commission review: April 1, 2016
- Proposed NPUF rule will be issued for public comment in the summer of 2016
  - 75-day comment period
  - All supporting documents will be issued together
  - NRC staff will hold a public meeting during the comment period
- Final NPUF rule should be issued early in 2018

NPUF Proposed Rule Summary

NPUF Proposed Rule Change	Class 103 Facilities	Class 104a Facilities	Class 104c Facilities	
	Commercial	Medical Therapy	R&D	Testing
1. Create a definition for NPUF	✓	✓	✓	✓
2. Eliminate license terms	N/A	✓	✓	N/A
3. Define the license renewal process	✓	N/A	N/A	✓
4. Require updated FSAR submittals	✓	✓	✓	✓
5. Amend timely renewal provision	✓	✓	✓	✓
6. Provide an accident dose criterion	✓	✓	✓	N/A
7. Extend applicability of 10 CFR 50.59	✓	✓	✓	✓
8. Clarify existing environmental reporting requirements	✓	✓	✓	✓
9. Eliminate NPUF financial qualification information requirement	✓	N/A	N/A	✓

Conclusion

- Eliminating licensing terms would reduce the burden on both the licensees and NRC staff
  - Allowed by the AEA
  - Minimum regulation on licensees
  - Continued oversight and inspections by NRC staff
  - Improved FSAR documentation
- Total quantitative benefit of the proposed rule: $13 million
- Maintain the safe operation of the facility while protecting the public health and safety

QUESTIONS?

BACK UP SLIDES

Regulatory Policy – Class 104

The policy for regulation of Class 104 NPUFs is described in the Atomic Energy Act of 1954, as amended, Section 104a. and c.

Sec. 104. Medical Therapy and Research and Development

a. …the Commission is directed to permit the widest amount of effective medical therapy possible with the amount of special nuclear material available for such purposes and to impose the minimum amount of regulation consistent with its obligations under this Act to promote the common defense and security and to protect the health and safety of the public.

c. The Commission is directed to impose only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread and diverse research and development.

Regulatory Policy – Class 103

The policy for regulation of Class 103 NPUFs is described in the Atomic Energy Act of 1954, as amended, Section 103.

Sec. 103. Commercial Licenses

a. The Commission is authorized to issue licenses to persons applying therefor to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use100 import, or export under the terms of an agreement for cooperation arranged pursuant to section 123, utilization or production facilities for industrial or commercial purposes.101 Such licenses shall be issued in accordance with the provisions of chapter 16 and subject to such conditions as the Commission may by rule or regulation establish to effectuate the purpose and provisions of this Act.

c. Each such license shall be issued for a specified period, as determined by the Commission, depending on the type of activity to be licensed, but not exceeding forty years from the authorization to commence operations and may be renewed upon the expiration of such period.

Regulatory Definitions

- Non-power reactor means a research or test reactor licensed under § § 50.21(c) or 50.22 of this part for research and development [*10 CFR 50.2 Definitions*].

- Research reactor means a nuclear reactor licensed by the Commission under the authority of subsection 104c of the Act and pursuant to the provisions of § 50.21(c) of this chapter for operation at a thermal power level of 10 megawatts or less, and which is not a testing facility as defined by paragraph (m) of this section [*§ 170.3 Definitions*].

Regulatory Definitions (cont.)

- Testing facility means a nuclear reactor which is of a type described in § 50.21(c) of this part and for which an application has been filed for a license authorizing operation at:

  (1) A thermal power level in excess of 10 megawatts; or

  (2) A thermal power level in excess of 1 megawatt, if the reactor is to contain:

  (i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or

  (ii) A liquid fuel loading; or

  (iii) An experimental facility in the core in excess of 16 square inches in cross-section. [§ 170.3 Definitions]

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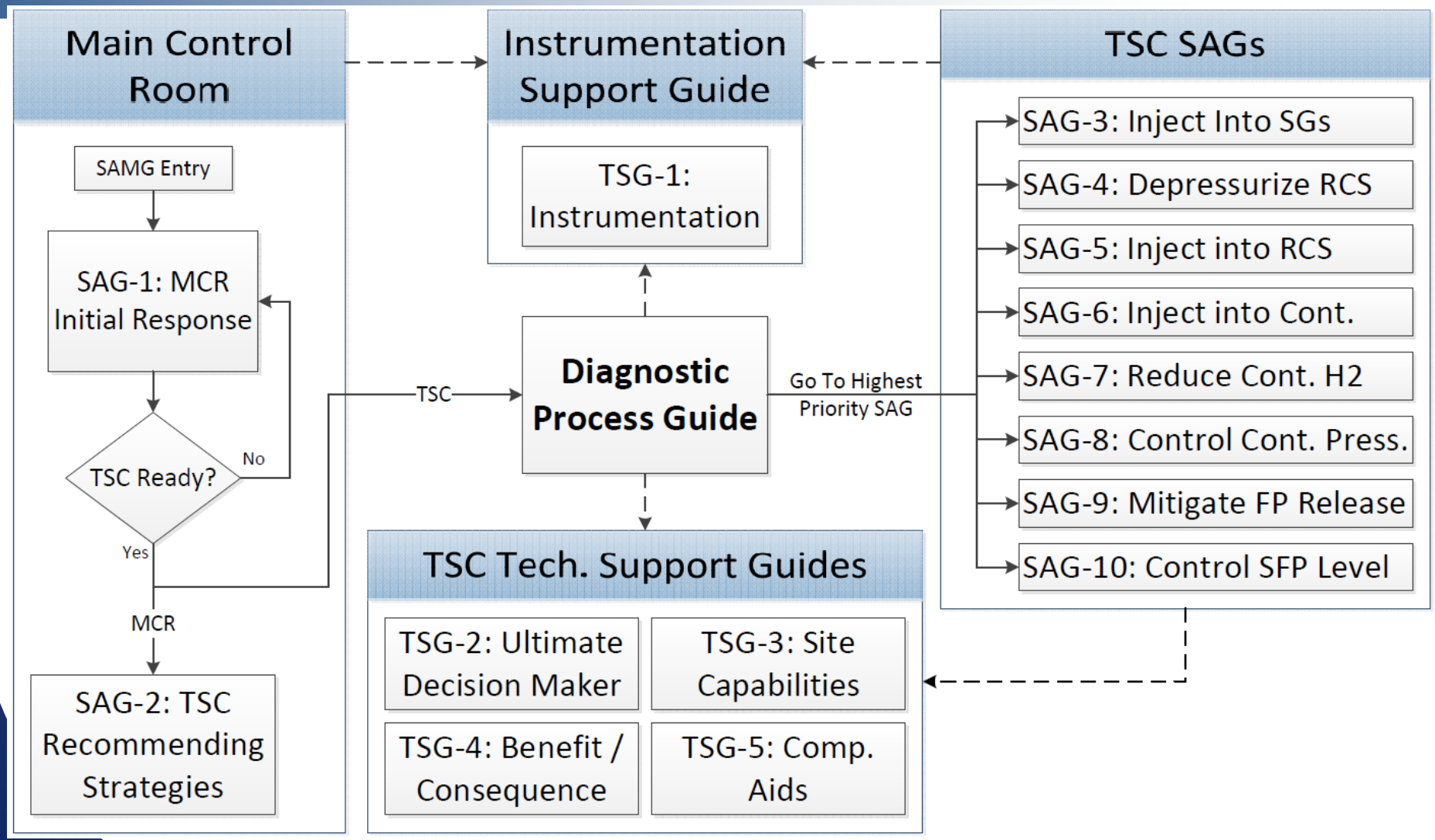
PWROG SAMG Instrumentation Guidance

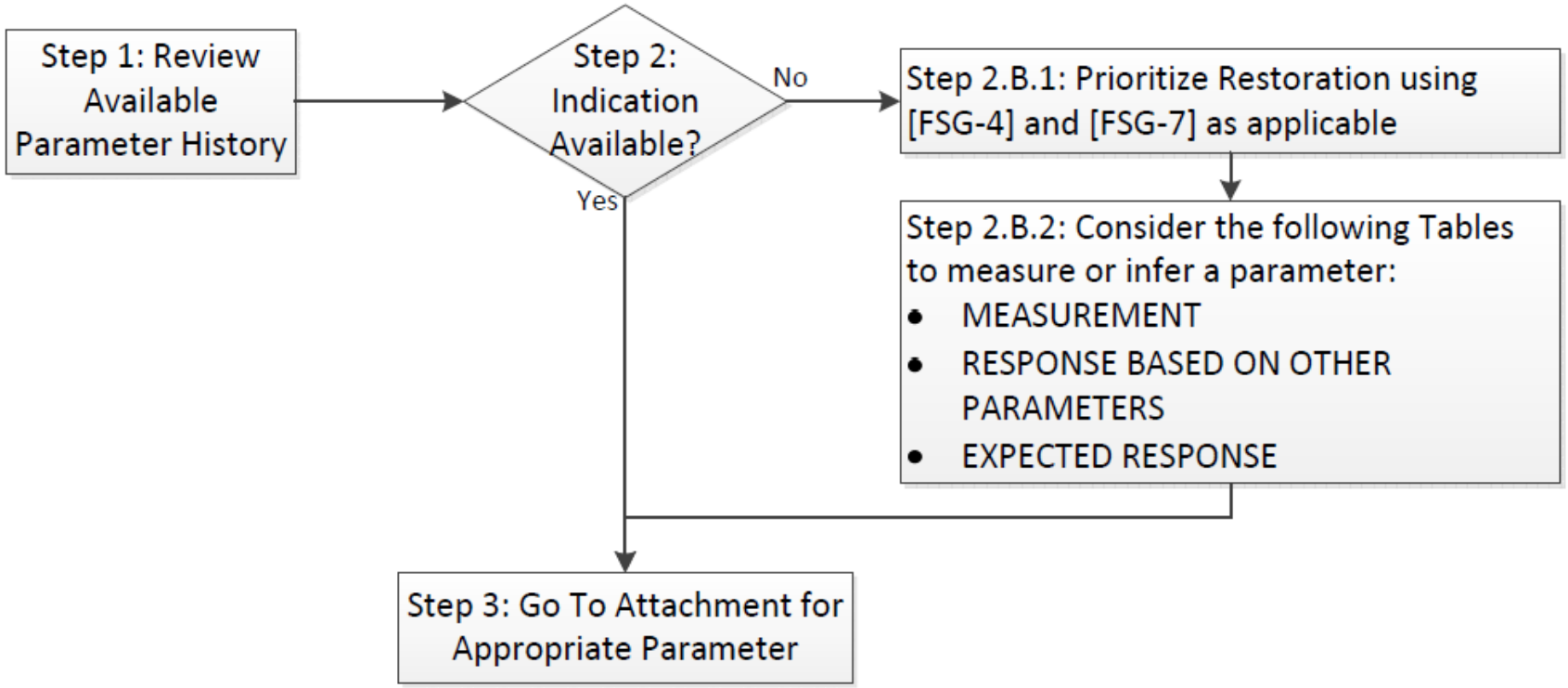
N. Reed LaBarge
Westinghouse Electric Co.
Pressurized Water Reactor Owners Group

PWROG SAMG Overview



TSG-1: Instrumentation

- TSG-1 provides additional guidance for identifying alternate means of measuring, inferring or confirming a plant parameter



Summary of TSG-1 Content

- TSG-1 Methods for Confirming Plant Parameters
  - Pre-calculated correction factors (if applicable)
  - Alternate confirmatory indications
  - Expected response based on other events or parameters
  - Instrument calibration details and measurement ranges
  - Severe accident instrumentation reliability information
- TSG-1 Methods to Measure / Infer Plant Parameters
  - Link to FSG-4 and FSG-7 for signal restoration and alternate measurement methods
  - Alternate means of locally measuring (including FLEX)
  - Expected response based on other events or parameters

SAMG Parameters in TSG-1

- TSG-1 includes guidance for all key SAMG plant parameters
- Additional parameters can be added during plant specific implementation

Parameter	Section
SG Water Level	Attachment A
SG Pressure	Attachment B
RCS Pressure	Attachment C
Reactor Vessel Water Level	Attachment D
RCS Injection Flow Rate	Attachment E
Core Temperature – In-Vessel	Attachment F
Core Temperature – Ex-vessel	Attachment G
Containment Water Level	Attachment H
Containment Pressure	Attachment I
Containment Temperature	Attachment J
Containment Hydrogen	Attachment K
Radiation	Attachment L
Spent Fuel Pool Level	Attachment M

Sample TSG-1 Info for SG Level

MEASUREMENT		
Primary	**Secondary**	**Other Indications**
SG Water Level (Wide Range)	SG Water Level (Narrow Range)	• Hand-held instrument at instrument rack • Moisture in steam line drains (very high SG level) • Excess moisture in extraction steam (very high SG level) • Steam in blowdown stream (very low SG level)

IMPACT OF ACCIDENT ON INSTRUMENTATION		
Instrument	**Reliability**	**Comments**
SG Water Level (Wide Range)	• Shows level over entire secondary side elevation • Likely to be useful during **ALL** phases of core damage • May need correction factors	• Calibrated at cold shutdown • WR NOT as accurate as NR
SG Water Level (Narrow Range)	• Only shows level above U-bend region • Likely to be useful during **ALL** phases of core damage • May need correction factors	• Calibrated at SG NOPT May need correction factors

Sample TSG-1 Info for SG Level

CORRECTION FACTORS – Response to Accident Conditions – *Wide Range SG*		
SG Pressure	Containment Conditions	
	DPG Orange Condition	DPG Red Condition
Normal Operating Conditions	Add (Note 1)%	Add (Note 4)%
Depressurized to 400 psig	Add (Note 2)%	Add (Note 5)%
Atmospheric Pressure	Add (Note 3)%	Add (Note 6)%
Notes 1 through 6 represent the correction to the indicated value for the conditions represented in the table. Example values have not been calculated in this response.		

CORRECTION FACTORS – Response to Accident Conditions – *Narrow Range SG*		
SG Pressure	Containment Conditions	
	DPG Orange Condition	DPG Red Condition
Normal Operating Conditions	Add (Note 7)%	Add (Note 10)%
Depressurized to 400 psig	Add (Note 8)%	Add (Note 11)%
Atmospheric Pressure	Add (Note 9)%	Add (Note 12)%
Notes 7 through 12 represent the correction to the indicated value for the conditions represented in the table. Example values have not been calculated in this response.		

Sample TSG-1 Info for SG Level

INDICATED and USEFUL RANGE			
Instrument	**Typical Range**	**Indicated Value**	**Useful Value**
SG Water Level (Wide Range)	• Lower tap is above SG tubesheet • Upper tap is at the top of the moisture separators	0-100% of the elevation BETWEEN the upper and lower taps	0-100% of the elevation BETWEEN the upper and lower taps
SG Water Level (Narrow Range)	• Lower tap is approx., the U-bend regions of the SG tubes • Upper tap is shared with wide range at the top of the moisture separators	0-100% of the elevation BETWEEN the upper and lower taps	0-100% of the elevation BETWEEN the upper and lower taps

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Closure of Fukushima Tier 2 and 3 Recommendations Related to:

Containment Vents, Hydrogen Control, and Enhanced Instrumentation

ACRS Full Committee Meeting

March 3, 2016

Japan Lessons Learned

Background

Fukushima Dai-ichi lessons learned developed and prioritized in a three-tiered approach (see SECY-11-0093 and SECY-11-0137)

Tier 3

- Require further staff study to support a regulatory action
- Requires completion of a shorter-term action to inform a longer-term action
- Dependent on availability of critical skill sets
- Dependent on the resolution of NTTF Recommendation 1

Tier 2

- Could not be initiated in the near term
- Need further technical assessment and alignment
- Depend on Tier 1 issues or availability of critical skill sets.
- Do not require long-term study

Tier 1

- Start without unnecessary delay
- Sufficient resource flexibility, including availability of critical skill sets

SECY-15-0137

- Some Tier 2 and 3 activities subsumed by Tier 1 activities
- Resolution plan for remaining Tier 2 and 3 activities provided in SECY 15-0137, “Proposed Plans For Resolving Open Fukushima Tier 2 and 3 Recommendations”
- Activities grouped into three bins:
  - Group #1 – Can be closed based on completed evaluations, progress made, and existing processes to address future work.
  - Group #2 – Sufficient information available and initial technical assessment complete; planned interactions with ACRS/external stakeholders; work to be completed and provided to the Commission (Information paper) by March 2016.
  - Group #3 – More detailed assessment and/or justification for resolution being prepared; ACRS/external stakeholder interactions planned; work to be completed in 2016.

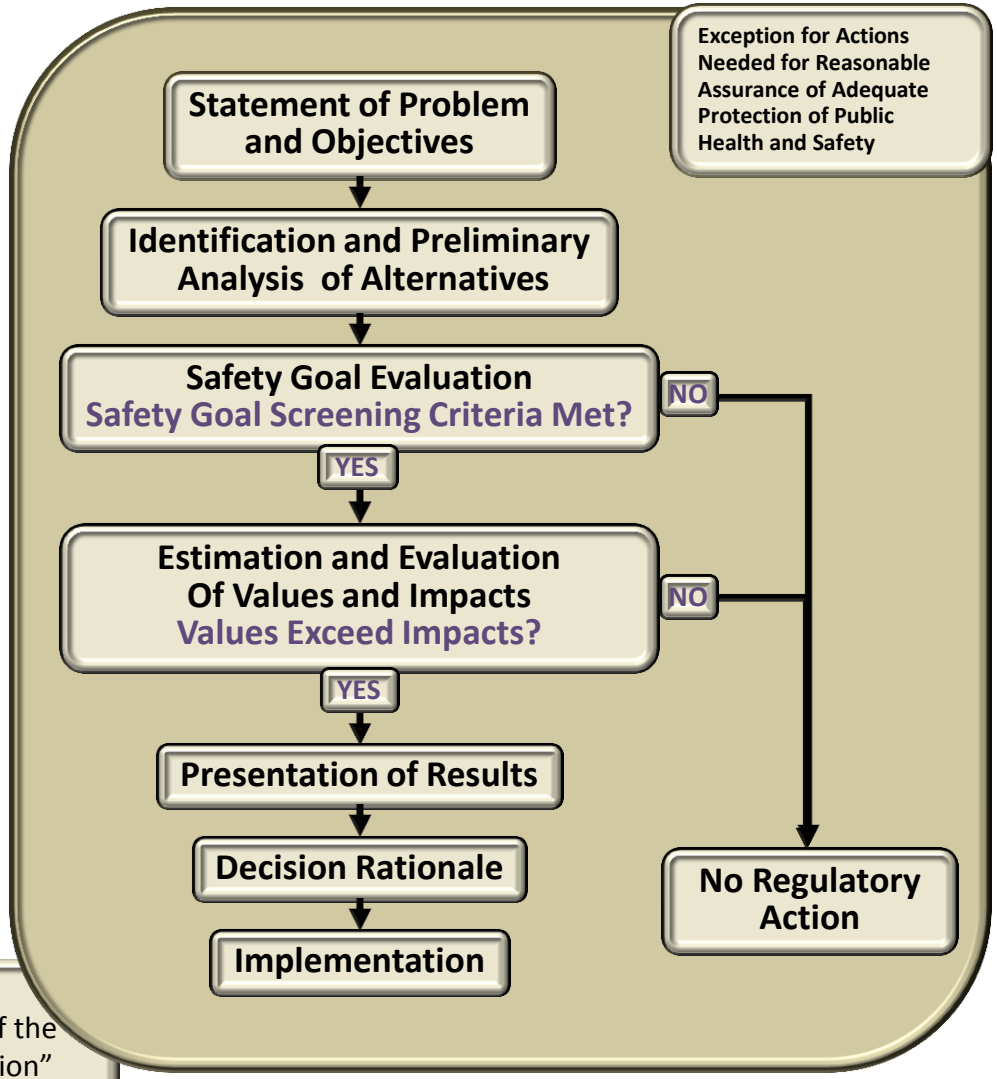
Background

- Purpose of meeting is to discuss Group 2 Recommendations
  - NTTF Recommendation 5.2 – Vents for Containment Designs Other Than BWR Mark I and Mark II
  - NTTF Recommendation 6 – Hydrogen Control and Mitigation
  - ACRS Recommendation – Reactor and Containment Instrumentation Enhancements for Beyond-Design-Basis Events
- Advisory Committee on Reactor Safeguards (ACRS) provided a November 16, 2015, letter report with its assessment of recommendations in SECY-15-0137
- Commission's Staff Requirements Memorandum for SECY-15-0137 dated February 8, 2016

Background

Evaluation of Possible Regulatory Actions

- NUREG/BR-0058
- 10 CFR 50.109 (Backfitting)
- Guidance from
  - Office of Management and Budget (OMB)
  - Government Accountability Office (GAO), and
  - Executive Orders



NUREG/BR-0058, Revision 4
"Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission"

White paper & March 2016 SECY paper

- Focus of evaluations remains whether possible regulatory requirements can be appropriately justified as required by NRC regulations
- Reflect Commission decisions such as the appropriate regulatory treatment of severe accident management guidelines (SAMGs)
- Acknowledge that various studies have identified possible enhancements
- Staff notes that work related to Fukushima and severe accidents will continue for many years
- Confirm initial findings that additional severe accident capabilities related to containments, hydrogen, and enhanced instrumentation do not warrant regulatory actions

Vents for Other Containment Designs (Group 2)

Initial Assessment (Oct 2015) included:

- Significant information available from previous activities and analyses (e.g. CPIP, NUREG-1150, SOARCA)
- Available technical information, including analyses for EA-13-109 and CPRR draft regulatory basis
- Discussion of related previous Commission decisions
- Evaluations for each containment type
- Initial conclusion: Further study is unlikely to demonstrate the need for regulatory action

Hydrogen Control and Mitigation (Group 2)

Initial Assessment (Oct 2015) included:

- Significant information available from previous activities and analyses
- Impact of existing regulations & mitigating strategies
- Insights from CPRR analyses, SOARCA, international initiatives, and previous Commission decisions
- Evaluations for each containment type
- Assessment of potential migration of hydrogen to reactor buildings or other structures
- Initial conclusion: Further study is unlikely to demonstrate the need for regulatory action

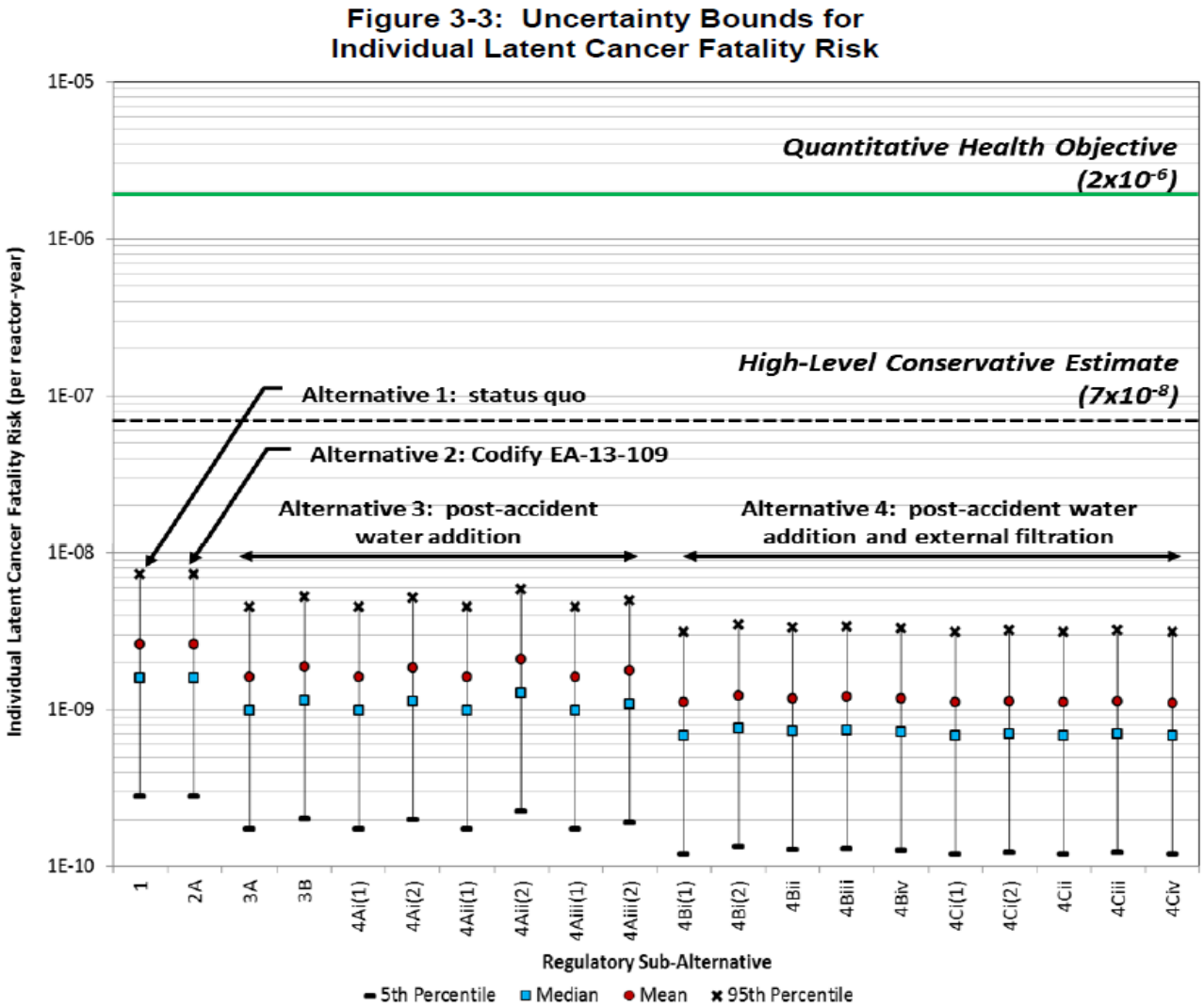
Evaluation of Other Containments and Hydrogen Control

Table 1
Recommendation 5.2 and 6 – Other Containment Designs and Hydrogen Control; Requirements and Practices

	Core Cooling Functions	Venting and/or Heat Removal for Containment Pressure Control		Other Containment Failure Modes and/or Core Debris Cooling	Release Reduction (Filtering)	Hydrogen Control	
		Pre-Core Damage	Severe Accident			Containment	Other
Mark I	EA-12-049 EA-13-109	EA-13-109 EA-12-049 EOPs FSGs	EA-13-109 SAMGs	EA-13-109 (CPRR)	N/A (CPRR)	EA-13-109 SAMGs	EA-13-109 SAMGs
Mark II	EA-12-049 EA-13-109	EA-13-109 EA-12-049 EOPs FSGs	EA-13-109 SAMGs	EA-13-109 (CPRR)	N/A (CPRR)	EA-13-109 SAMGs	EA-13-109 SAMGs
Mark III	EA-12-049	EA-12-049 EOPs FSGs	SAMGs	SAMGs	N/A (current assessment)	GSI-189 EA-12-049 SAMGs FSGs	GSI-189 EA-12-049 SAMGs FSGs
Ice Condenser	n/a	EOPs	SAMGs	SAMGs	N/A (current assessment)	GSI-189 EA-12-049 SAMGs FSGs	GSI-189 EA-12-049 SAMGs FSGs
Large Dry	n/a	EOPs	SAMGs	SAMGs	N/A (current assessment)	SAMGs	N/A (current assessment)

EA-12-049: Mitigation Strategies Order
EOPs: Emergency Operating Procedures
SAMGs: Severe accident management guidelines
EA-13-109: BWR Mark I/II Severe accident capable vent order
FSGs: FLEX (Mitigating Strategies) Support Guidelines
GSI-189: Generic Safety Issue re: Hydrogen Issues

Mark I/II Containments
(Order EA-13-109 & CPRR Rulemaking)

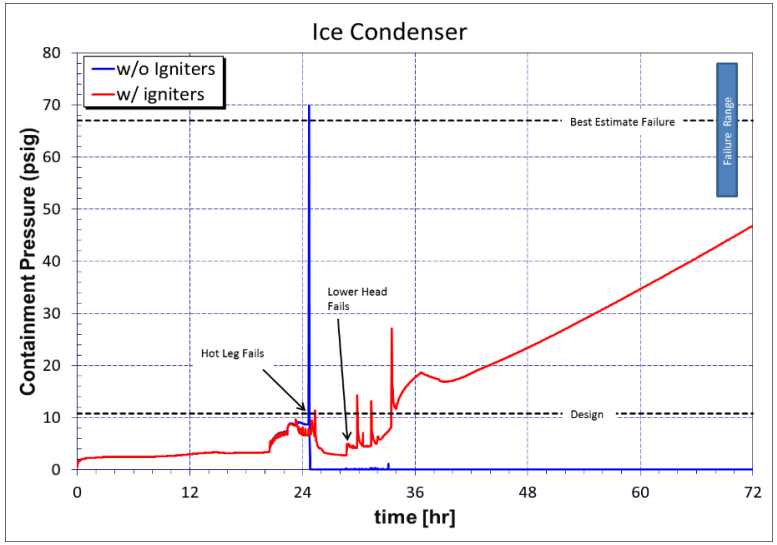


Japan Lessons Learned

Vents for Other Containment Designs and Hydrogen Control and Mitigation

- ACRS letter report comments:
  - Reserves recommendation until after review of staff's additional evaluation
  - Pathways for hydrogen release to the reactor building under severe accident conditions should be considered
  - Provided expectations that NRC staff will maintain research programs and will monitor international activities in this area
- White Paper (February 2016) added:
  - More recent analyses to support technical justification
  - Additional discussion of regulatory evaluation
    - Insights from high-level conservative estimate for Mark I and Mark II
    - Additional discussion of containment penetrations and containment conditions (pressure/temperature)

Vents for Other Containment Designs



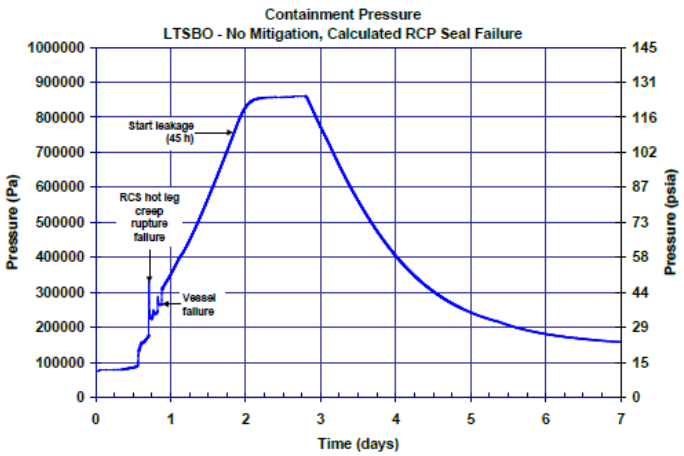
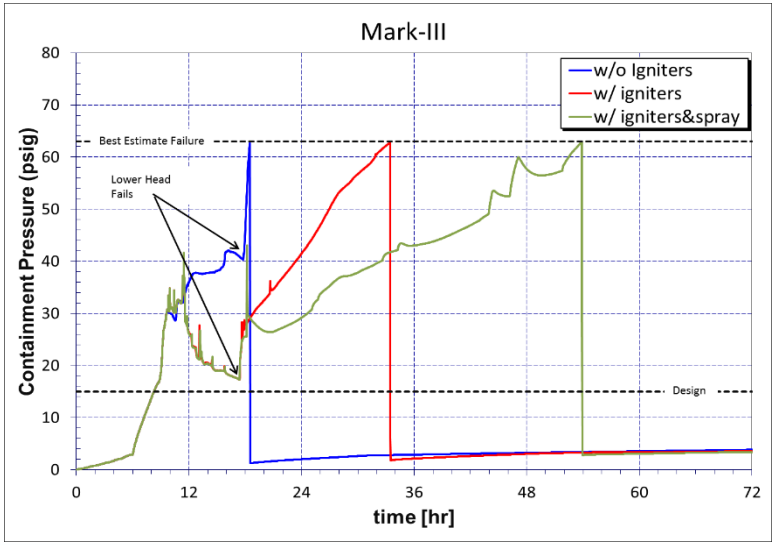
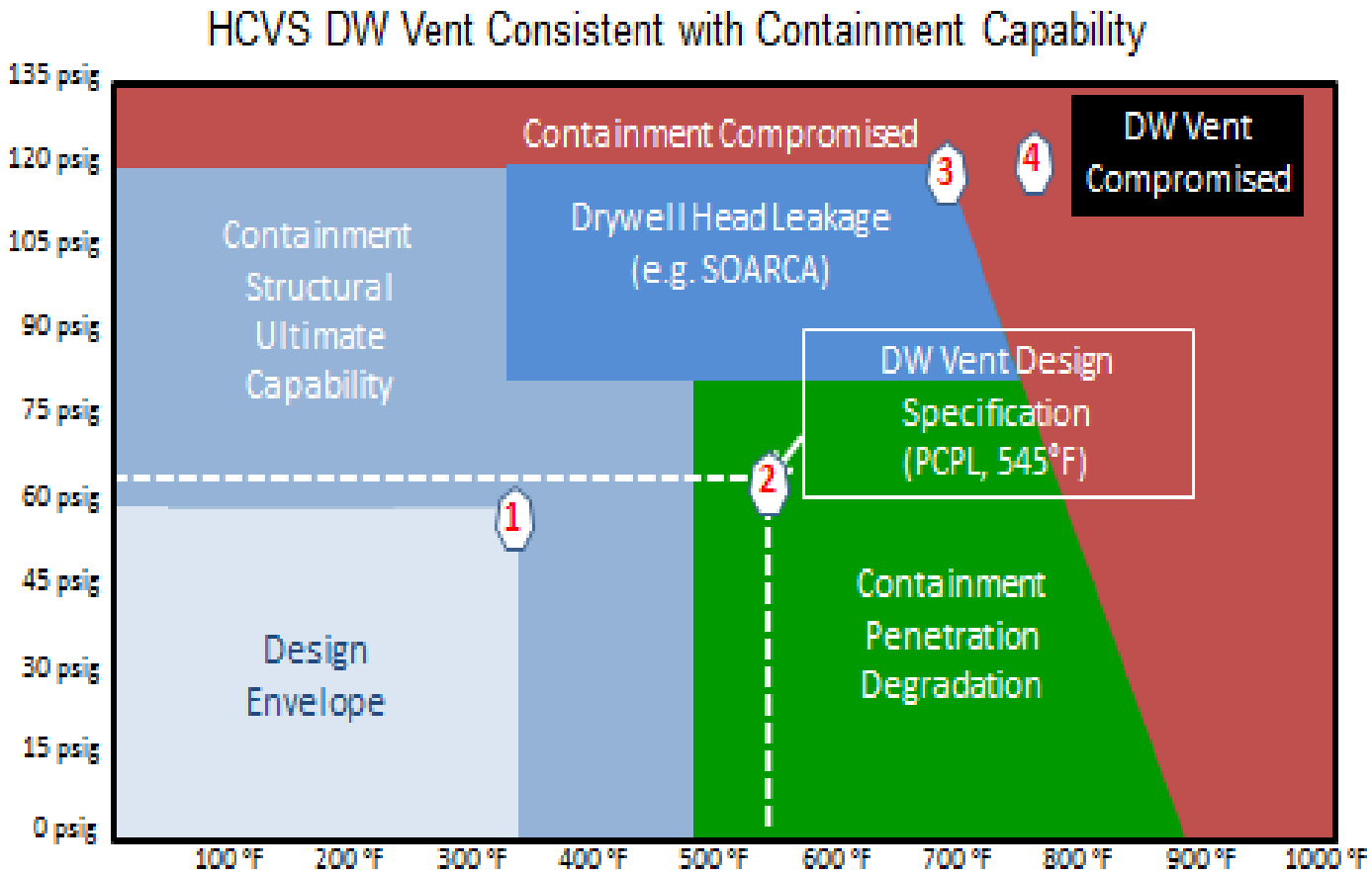


Figure 5-5 Unmitigated long-term station blackout containment pressure history



Hydrogen Control and Mitigation

Supporting Evaluation for NEI 13-02 (Mark I/II Vent Order)



Containment Vents and Hydrogen Control and Mitigation

- Summary:
  - Staff has not performed new plant simulations or risk evaluations for each containment type comparable to that done for Mark I and II designs
  - Staff reviewed available studies and performed some additional evaluations, including work being prepared for ice condenser SOARCA study
  - Reflects benefits of improved capabilities for hydrogen igniters (Mark III and Ice Condenser containments)
  - Conclude that significant margins exist between estimated plant risks that might be influenced by improvements to containment performance and the NRC established safety goals

Development of SECY Paper (Planned changes from White Paper)

- Enhancements Being Made in Response to ACRS Subcommittee Discussions
  - Explanation of QHO as aggregate measure of risk and regulatory evaluations as looking at delta-risks from possible plant changes
  - Additional information from recent SOARCA study for large-dry containments (Surry) related to containment performance and hydrogen
  - Additional information related to short-term station blackout

Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions

SECY-15-0137 assessment included:

- Existing requirements for protection of instrumentation
- Insights gained from proposed rulemaking for mitigation of beyond-design-basis events
- A discussion on continued work on endorsement of industry standards (to be used voluntarily)
- Interactions with domestic and international organizations
- A discussion on continued reviews on new reactor equipment survivability in accordance with policy
- Initial conclusion: Limited additional safety benefit; no further regulatory action needed

Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions

NEI 12-06, Mitigating Strategies
Typical Parameters to Remain Powered During Load Shed

PWRs [Pressurized Water Reactors]	***BWRs [Boiling Water Reactors]***
• *SG [Steam Generator] Level* • *SG Pressure* • *RCS [Reactor Coolant System] Pressure* • *RCS Temperature* • *Containment Pressure* • *SFP Level*	• *RPV [Reactor Pressure Vessel] Level* • *RPV Pressure* • *Containment Pressure* • *Suppression Pool Level* • *Suppression Pool Temperature* • *SFP Level*

The plant-specific evaluation may identify additional parameters that are needed in order to support key actions identified in the plant procedures/guidance (e.g., isolation condenser (IC) level), or to indicate imminent or actual core damage.

Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions

ACRS letter report comments:

- Expand discussion of calculational aids that could be used to supplement or replace data instruments when required
- Include discussion of how severe accident management guidelines and calculational aides are capable of leading operators to take the correct actions, even if minimal instrumentation is available or the indications are suspect
- Provide discussion of validation work used to determine what instrumentation is necessary before during and subsequent to a severe accident

Reactor and Containment Instrumentation for Beyond-Design-Basis Conditions

White Paper (February 2016) added:

- Status of revision to IEEE-497, “Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations”
- Status of revision to Regulatory Guide 1.97, “Criteria for Accident Monitoring Instrumentation”
- Expanded discussion of SAMG instrumentation guidance to address ACRS comments
- Input from external stakeholders

Instrumentation and SAMGs

- Control parameter assessment guideline
  - Validate control parameter readings
  - Monitor and trend the important parameters
  - Adapt frequency of monitoring based on rate of change of monitored parameter
  - Perform technical support guidance calculations to support parameter assessment
- Plant status assessment guideline
- Function status assessment guideline
- EPG/SAG action assessment guideline

Instrumentation and SAMGs

- Principles used when validating instrumentation:
  - Use of other indications (indirect measures) to confirm readings, to ensure that assessments are confirmed from two or three sources
  - When actions are taken, parameters are monitored to ensure that instruments respond as would be expected
  - Trends and changes in trends may be true even when absolute values or readings are not
  - Assessment to consider physical explanations for instrument readings

Development of SECY Paper (Planned changes from White Paper)

- Enhancements Being Made in Response to ACRS Subcommittee Discussions
  - Additional discussion of technical support guidance, with focus on computational aids
  - Both SAMGs and NRC incident response tools provide aids to assist personnel performing core damage assessments
  - Reference to NRC incident response guidance in NUREG/BR-0150, Response Technical Manual 96 (RTM-96), which includes similar aids
  - SAMGs and supporting instrumentation and calculational aids serve high-level goal of restoring or maintaining cooling of core debris and containment integrity

Questions & Discussion