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526th Meeting

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

October 6, 2005

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This transcript has not been reviewed, corrected and edited and it may contain inaccuracies.

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

526TH MEETING

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THURSDAY,

OCTOBER 6, 2005

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The meeting came to order at 8:30 a.m. in room
T2-B3 of Two White Flint North, Rockville, Maryland.

William J. Shack, Vice Chairman, presiding.

PRESENT:

WILLIAM J. SHACK, VICE CHAIR

DANA A. POWERS, MEMBER

VICTOR H. RANSOM, MEMBER

MARIO V. BONACA, MEMBER

GEORGE E. APOSTOLAKIS, MEMBER

RICHARD S. DENNING, MEMBER

THOMAS S. KRESS, MEMBER

JOHN D. SIEBER, MEMBER AT LARGE

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1 ALSO PRESENT:
2 JOHN T. LARKINS, DESIGNATED FEDERAL OFFICIAL
3 ASHOK C. THADANI, DEPUTY EXECUTIVE DIRECTOR,
4 SAM DURAISWAMY, STAFF
5 MICHAEL L. SCOTT, STAFF
6 JENNY M. GALLO, STAFF

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P R O C E E D I N G S

8:32 A.M.

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3 VICE CHAIRMAN SHACK: The meeting will now
4 come to order. This is the first day of the 526th
5 meeting of the Advisory Committee on Reactor
6 Safeguards. During today's meeting, the Committee
7 will consider the following: the interim review of
8 the license renewal application for the Browns Ferry
9 Nuclear Plant, Units 1, 2 and 3; proposed
10 recommendations for resolving Generic Safety Issue 80,
11 pipe break effects on control rod drive, hydraulic
12 lines and the dry wells of boiling water reactor Mark
13 1 and 2 containments; resolution of ACRS comments on
14 the draft final regulatory guide; risk-informed
15 performance-based fire protection for existing
16 lightwater reactor nuclear power plants; Davis-Besse
17 reactor vessel head integrity calculations; quality
18 assessment of selected NRC research programs; and
19 preparation of ACRS reports.

20 This meeting is being conducted in
21 accordance with provisions of the Federal Advisory
22 Committee Act. Dr. John T. Larkins is the Designated
23 Federal Official for the initial portion of the
24 meeting.

25 We have received no written comments or

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1 request for time to make oral statements from members
2 of the public regarding today's session. A transcript
3 of portions of the meeting is being kept and it is
4 requested that speakers use one of the microphones,
5 identify themselves and speak with sufficient clarity
6 and volume so they can be readily heard.

7 As you will note, I'm not Graham Wallis,
8 Chairman of the ACRS, who is still in the south of
9 France somewhere. So --

10 MEMBER APOSTOLAKIS: Who are you?

11 (Laughter.)

12 VICE CHAIRMAN SHACK: Some items of
13 current interest, if you look in your package, you'll
14 see a yellow announcement that will describe some of
15 the reorganization that's occurred in NRR. There's
16 also an article that describes Chairman Diaz' multi-
17 design initiative on international certification of
18 reactors. And again, a number of other speeches and
19 items of interest from the other Commissioners.

20 I do want to introduce Gabe Taylor. As of
21 October 3rd, Gabe began a six-month rotation with the
22 ACRS. During this rotation, Gabe will assist the
23 Committee with its review of the digital INC research
24 plan and the ESBWR design. Gabe joined the NRC in
25 April of 2005 as a general engineer participating in

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1 the Nuclear Safety Professional Development Program.
2 He graduated from Penn State University with a
3 Bachelor of Science degree in Electrical Engineering
4 with a focus on power and control system design.
5 During his first six months in the Agency, Gabe worked
6 in the Office of Nuclear Reactor Regulation.

7 One other thing I wanted to mention, as I
8 got an email last night that told me that Spence Bush
9 had passed away on October 2nd. Spence is a former
10 Member of the Committee, Chairman of the Committee and
11 one of the last of the generation of the real nuke
12 founders. Spence told me once he was driving, he
13 drove Oppenheimer to the Trinity site in the Jeep, so
14 he goes back all the way to Day 1 of the nuclear era.
15 He was a remarkable man. He always sort of struck me
16 as the Energizer Bunny. He was about so high and just
17 sort of kept on going, all the time.

18 Our first item of interest today is Browns
19 Ferry Nuclear Plant license renewal application and
20 Mario will lead us through that.

21 MEMBER BONACA: Okay, good morning.
22 Yesterday, the Plant License Renewal Subcommittee met
23 to review the interim SCR for the Browns Ferry Nuclear
24 Plant, Units 1, 2 and 3 license renewal. We also met
25 on September 21st and previously in the month of

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1 August at the Browns Ferry to familiarize ourselves
 2 with this complex application or a number of
 3 applications that the Browns Ferry units are going
 4 through right now. You are familiar with the fact
 5 that Unit 1 is still at the end of its 22 years almost
 6 of layout conditions and will be starting in 2007.
 7 That restart will include an EPU of 20 percent and
 8 although the EPU is not part of the consideration for
 9 license renewal, I raise this issue because of the
 10 complexity of the application and the fact that Unit
 11 1 does not have the expected operating experience that
 12 the rule intends to have as stated in the Statement of
 13 Consideration.

14 So yesterday, during our meeting we
 15 discussed a number of issues which I believe I would
 16 like to just briefly summarize that should be of the
 17 interest to the Committee today.

18 The first one is how do you deal with the
 19 issue of operating experience of the licensee. The
 20 licensee has assumed that the operating experience for
 21 Unit 1, 2 and 3 is applicable to Unit 1. We have
 22 raised the issue of proficiency. Clearly, it is
 23 applicable, like general experience or operating
 24 experience, but is it sufficient, particularly for
 25 dealing with components which were in lay-up and may

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1 have latent aging effects that will only surface after
2 the plant is in operation.

3 So we discussed that issue and we felt
4 that there should be in the SER a comprehensive
5 discussion of this issue up front. This is being
6 recognized both by the licensee and the NRC staff and
7 they have agreed to in the final SER to incorporate
8 such a discussion.

9 The second issue is the fact that the
10 licensee has committed periodic inspection of separate
11 components that were in lay-up and this is really
12 essentially a compensatory action for the lack of
13 operating experience for those components. And I
14 think that we were favorably impressed by that
15 program, although the program is not sufficiently
16 defined. We heard a number of commitments on the part
17 of the licensee and we'd like you to hear today
18 because those commitments are important to determine
19 whether or not, in fact, the operating experience with
20 this compensating factors is adequate for Unit 1.

21 As a result of the discussion on the
22 periodic inspections, the staff decided to issue a new
23 open item to the licensee dealing with this very issue
24 which means the expectation there will be a program
25 defined by the time the SER, the final SER is issued

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1 that will address this very issue of the periodic
2 inspection program and what it will consist of in
3 detail.

4 Other comments that we had had to do with
5 the fact that the application in many ways, of the
6 schedules of the plant, from the moment it reached --
7 the license renewal was submitted in 2003 to today,
8 has changed significantly because the plant is being
9 refurbished before the start. Therefore, the SER
10 seems to not provide an evaluation on a fixed status
11 of the plant, but there are changes in status of the
12 plant that are being addressed within it and the most
13 uncommon that there should be some better
14 understanding of what plant we are talking about in
15 the SER and maybe in the application.

16 With that I will turn now to Dr. Kuo. I
17 understand there will be first of all a presentation
18 by Browns Ferry and then the staff will address the
19 SER.

20 Dr. Kuo.

21 MR. KUO: Thank you and good morning, Dr.
22 Bonaca. I'm the Program Director for License Renewal
23 and Involuntary Impacts Program. To my right are the
24 project managers for this review: Ram Subbaratnam and
25 Yoira Diaz.

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1 Yesterday, as Dr. Bonaca reported, we had
2 a meeting, a supplemental meeting on the review of
3 Browns Ferry license renewal application and I just
4 want to make it clear that we originally had in the
5 SER two open items, but as a result of yesterday's
6 meeting, that number has increased to four. As Dr.
7 Bonaca mentioned one is the result of ACRS' review on
8 the periodic inspection and the other is the result of
9 a regional inspection.

10 And also, I want to repeat what I said
11 yesterday that this review is rather complicated than
12 usual. The complexity comes from the three concurrent
13 actions like Dr. Bonaca just mentioned. First one is
14 a Unit 1 restart and second one is the license
15 renewal. The third one is EPU. All of these three
16 actions are being carried out concurrently and that
17 adds to some of the complexity to this review, but
18 this was clearly described in our SER. Our focus in
19 this review is to review the license renewal
20 application at the current power level, not at the EPU
21 level. That is the one major thing that we want to
22 make it very clear to this Committee and whatever the
23 impact from EPU review will be, that will be taken
24 care of in the time of EPU.

25 And also, I just want to mention that I

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1 have Frank Galliespie in the audience who is our
2 Deputy Director for the Division of Program
3 Improvements, but after October 30th, Mr. Galliespie
4 is going to be the Director of the License Renewal
5 Division.

6 Frank do you want to say something? If
7 Frank doesn't have any opening comments, then we will
8 go ahead with the review, turning this over to the
9 Applicant.

10 MR. CROUCH: Good morning. My name is
11 Bill Crouch. I'm the site licensing manager at Browns
12 Ferry Nuclear Plant. We appreciate the opportunity to
13 come and talk to you today. Some of you we got to
14 talk to yesterday and others, this may be your first
15 time of hearing some of this story. Others it may be
16 the second because you may have been with us down at
17 Browns Ferry back in August. But we appreciate the
18 opportunity to come and talk to you and tell you that
19 a little bit of the details about our license renewal
20 project.

21 In addition to myself, we have several
22 members of our Browns Ferry staff here today. I'm not
23 going to introduce all of them to you, but I'll tell
24 you some of the key players. We have Rich DeLong who
25 is our Site Engineering Manager. He has

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1 responsibility for all of the engineering activities
2 on site. He is the overall program owner for license
3 renewal. He owns it. He says, as he said yesterday,
4 it is mine. He's got it. That means that he and his
5 staff own this program and they understand the
6 importance of it.

7 We also have with us today Ken Brune, who
8 is the Project Manager over the license renewal
9 project. He and his staff are here today so that they
10 can answer any kind of technical questions.

11 We also have Joe Valente here with us.
12 Joe is the Unit 1 Engineering Manager. He has his
13 staff with him here also. So we can answer questions
14 about Units 1, 2 or 3 or the recovery of Unit 1 or
15 license renewal for any of those. We appreciate the
16 opportunity to answer any questions you have.

17 As I said, yesterday, we made a
18 presentation to the staff and we're going to use the
19 same package today as what we had yesterday. I'm
20 going to give you a shortened version of it and so I
21 will be telling you, we will now move on to page so
22 and so. We'll be skipping some pages and some bullets
23 and things along the way. We wanted you to have the
24 full information, the full packet in case you wanted
25 to see it. So that's how we're going to proceed from

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1 here.

2 So we'll be starting on page 2 of the
3 packet that you've got in front of you there. We
4 recognize that there are three big issues before us,
5 the restart of Unit 1, I think being the biggest; and
6 then we've got license renewal and EPU. We realize
7 that there are interrelationships, close
8 interrelationships between license renewal and EPU and
9 we've been considering that all along.

10 But we also, as we started through this
11 process, we talked to the staff and we recognized that
12 when we submitted the license renewal application that
13 we had to make the license renewal application at
14 current license thermal power since the EPU had not
15 been approved yet. The reason for that was that if we
16 submitted license renewal at EPU conditions, once it
17 was approved by the NRC, that was an implicit approval
18 of EPU, if it was written that way. So instead, the
19 license renewal application is written for current
20 license thermal power.

21 As we go through this process of reviewing
22 the three big issues, we will always, as part of our
23 design process, construction process, etcetera, we
24 will consider the other aspects of this, but when we
25 talk about the review process, you have to consider

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1 them one at a time and as you move to the next one,
2 you look back at the one that you've already approved
3 and make sure that the effects have been considered.
4 So it's a backward looking type process as we go
5 along, but we've included in our processes, as we've
6 been doing the recovery efforts.

7 As far as Browns Ferry, Browns Ferry --
8 there's three units. They're all GEBWR-4 units with
9 Mark 1 containments. They're all in a common
10 building. They were originally designed and
11 constructed by PVA. They were designed and
12 constructed to be essentially identical units. Two of
13 them are opposite hand, but other than that they are
14 operationally identical. When they were built, they
15 had all the same equipment, the same materials of
16 construction, etcetera, etcetera. So they were the
17 same. As it shows up there, the approximate years of
18 operation is in calendar years.

19 Everybody is probably aware of some of the
20 history of Browns Ferry and that we operated for a
21 while and then we shut down. Units 2 and 3 have been
22 operating as shown up there since 1991 and 1995,
23 respectively. Unit 1 has been in extended lay-up
24 condition since 1985 and we're in the process of
25 recovering that plan right now from May of 2007.

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1 Once we go through and do the restart
2 activities, Unit 1 will be operationally identical to
3 Units 2 and 3.

4 MEMBER POWERS: Why was Unit 1 laid up?

5 MR. CROUCH: Why was it laid up or why was
6 it shut down?

7 MEMBER POWERS: Shut down.

8 MR. CROUCH: It was shut down in 1985 due
9 to management and safety concerns that we had not come
10 into conformance with various regulations such as
11 Appendix R, EQ, a lot litany of things, and also
12 perceived management weaknesses. We shut all three
13 units of Browns Ferry down as well as units at
14 Sequoia. At that point in time, we negotiated with
15 the NRC a plan for recovery and it was laid out in
16 what's called the Nuclear Performance Plan, three
17 volumes, accepted. It went through and gave us
18 updates for how we needed to revise our management
19 team, what we needed to do for our processes and then
20 specific technical issues that had to be addressed.

21 MEMBER POWERS: You apparently addressed
22 those for Units 2 and 3, but not for Unit 1?

23 MR. CROUCH: At that time, we did
24 obviously the management changes applied to the whole
25 utility and they were for the whole site, so yes, they

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1 were done in that respect for Unit 1. The process
2 changes were done for Unit 1 at that time, but we did
3 not do the technical programmatic or technical
4 configuration type changes at that time.

5 MEMBER POWERS: Why not?

6 MR. CROUCH: It was a staged recovery, so
7 we did one unit first and then we moved on to the next
8 unit. Once we got Unit 3 recovered, at that point in
9 time we did not need the power, so we did not
10 immediately proceed.

11 MEMBER POWERS: There must have been some
12 reason to do 2, 3 and then eventually 1.

13 MR. CROUCH: It was based upon which unit
14 we believed was in the best condition to be recovered
15 the fastest.

16 MEMBER POWERS: So somehow Unit 1 was in
17 a worse condition than the others?

18 MR. CROUCH: It had less of the older mods
19 done to it. There would have been more work to get it
20 back running.

21 MEMBER POWERS: Thank you.

22 MR. CROUCH: As I said, once we recover
23 Unit 1, Unit 1 will be operationally identical to
24 Units 2 and 3. And I want to make sure everybody
25 understands what we mean by operationally identical.

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1 As we recovered units 2 and 3, we installed hardware
2 that was available at the time. Here we are 10 years
3 or more later and some things you just physically
4 cannot buy any more. Companies have gone out of
5 business or technology has changed. For example,
6 recorders in the control room will not longer be paper
7 recorders like we have on 2 and 3. They're paperless
8 recorders. They're electronic. But as far as the
9 operator is concerned, it's still a recorder. It
10 still supplies the same information to him. You go
11 out to the plant, into the more hardware, the piping
12 systems, you'll find cases where valve manufacturers
13 have gone out of business. It used to be a Brand X
14 gate valve, well, you can't buy a Brand X gate valve
15 any more, so we had bought a Brand Y gate valve. It's
16 still a gate valve. It's still the same size, same
17 material, everything. It's just a different brand.

18 MEMBER POWERS: All gate valves have
19 exactly the same reliability?F

20 MR. CROUCH: We have bought currently the
21 best valves. We have bought the valves that have high
22 reliability, whether they have exactly the same I
23 would hope everything we bought by current day
24 standards is as good or better than what was bought
25 back in 1991 and 1995.

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1 So the units will not be, if you walked
2 out there, completely identical from the standpoint of
3 brand names and stuff like that, but from an
4 operational standpoint, they will be the same. We've
5 used the same materials with the same general
6 configuration as far as having a gate valve where a
7 gate valve is supposed to be, etcetera.

8 We'll now turn to page 3 of the
9 presentation. For license renewal, this was submitted
10 as a three-unit application. As we started the
11 license renewal process, we had not started the Unit
12 1 recovery at the time, so when we internally started
13 the application, it was to be a two-unit application.
14 As we decide to restart Unit 1, we then backed up and
15 made it a three-unit application. So it does cover
16 all three units.

17 The application recognizes that Unit 1 is
18 in recovery status and we'll talk about that down
19 through the course of the slides here.

20 You can see there the current license
21 expiration dates. The license renewal application is
22 based upon current license thermal power. As we
23 talked about we could not reference it to a future
24 power, because that would be an implicit approval of
25 that power level, so for Unit 1 it's based upon the

1 original license thermal power of 3293. For Unit 2
2 and 3, it's based upon the current license thermal
3 power which is 105 percent of original or 3458
4 megawatts thermal.

5 It was recognized that Unit 1 was in a
6 recovery process and there was lots of modifications
7 to be made to bring it into conformance with Units 2
8 and 3 from an operational standpoint. As we started
9 the license renewal process, TVA and NRC staff went
10 through and jointly figured out which of these various
11 modifications were pertinent to license renewal and in
12 the course of the application, there is an appendix to
13 the application, Appendix F or Appendix Foxtrot, that
14 lists 13 major programs or modifications that will
15 bring the two units into conformance. These things
16 are such things as replacing the IGSCC with acceptable
17 piping for recirc. RWCU. They're adding things in
18 like hardened wet well vent, the alternate leakage
19 treatment path for MSIV leakage. And there's 13 of
20 them.

21 So once those 13 items are implemented,
22 then the two units will be back in operational
23 fidelity for the purpose of license renewal.

24 We'll move on to page 4 now. As we
25 started through the license renewal process, we did

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1 the scoping of the systems that were involved. We
2 used our licensing basis documents. We also used the
3 documents that apply to specific regulated events such
4 as Appendix R, EQ Atlas, etcetera. Based upon that,
5 we came up with 77 mechanical and electrical systems
6 that were within the scope of license renewal
7 projects. Those were laid out. They're marked up on
8 drawings, color coded, so we know exactly what's in
9 scope and we use that as a basis for our license
10 renewal activities.

11 Moving on to page 5, after we had the
12 scoping done, we went through our various time limited
13 aging analyses. The various ones are shown up there.
14 I won't go through any of them in particular, unless
15 you want some particular details on them.

16 Moving on to page 7, as a result of our
17 license renewal application, we determined that we
18 needed 39 aging management programs. Of these, 38 of
19 them are common to Units 1, 2 and 3. And there's one
20 that's specific to Unit 1 and that's our Unit 1
21 periodic inspection program.

22 The next four pages of your package list
23 those programs and I'm not going to go over each and
24 every one of them, but they're listed there. They're
25 broken up into three categories: those that were just

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1 okay, exactly as is; those that required some
2 enhancement to make them technically sound for license
3 renewal; and those that required some enlargement,
4 basically to include the scope of Unit 1.

5 There's also six brand new aging
6 management programs that are listed on the fourth page
7 there.

8 As you will hear through the course today,
9 as we went through this process, the region came in
10 and did an inspection of our programs back in December
11 and we were not ready for that at the time. We had
12 not really started the aging management programs at
13 the time. Since that time, we have gone through and
14 developed all these programs. They are marked up with
15 procedures. They are permanently stored. They will
16 be implemented into the procedures as we get closer to
17 the license renewal process, for those that aren't
18 currently in there already.

19 It is a track process, controlled under
20 our Corrective Action Program to ensure that the
21 changes get into procedures.

22 The overall program, as I said earlier, is
23 owned by Rich DeLong and the site engineering
24 contingency there. They are actively involved in the
25 review of these programs. As these programs are being

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1 developed, they did a technical review on all of them
2 to make sure that they were technically sound, that
3 they met the requirements of the engineering aging
4 what we've learned document, and regarding these
5 programs, we'll be implementing them over the course
6 of time.

7 Move on to page 13. As I talked about,
8 there was one unique program for Unit 1. It was
9 recognized that there's a large amount of equipment
10 out in Unit 1 that is being physically replaced as
11 part of the recovery. This will be brand new
12 equipment, brand new piping, brand new valves, brand
13 new cabling, etcetera. It was also recognized that
14 there was still a substantial portion of Unit 1 that
15 as not being replaced. It will be the original
16 equipment that is still being used.

17 We were confident that this equipment
18 would be good for the period of current operation, as
19 well as the extended operation, however, we wanted to
20 make sure that the equipment was not experiencing or
21 exhibiting any type of aging mechanism that we were
22 not aware of. So in order to ensure this, we will do
23 additional inspections on the non-replaced equipment
24 out in the plant to ensure that we know what's going
25 on out there.

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1 What we will do is there will be
2 inspections performed prior to restart that will
3 provide us a baseline set of information. We will
4 then do another inspection after several years after
5 restart to see if there's any degradation occurring.
6 And once we enter the period of extended operation, we
7 will do another set of inspections and based upon the
8 results of those three inspections, we will decide if
9 there's anything unusual happening in Unit 1 or if
10 there's any effects coming from the lay-up that we
11 were not aware of.

12 So this gives us confidence that we will
13 know the condition of Unit 1 as we proceed into the
14 periods of extended operation.

15 MEMBER POWERS: Perhaps this isn't defined
16 too well yet, but what's going to be different about
17 these periodic inspections that will be different from
18 the other inspections that you're going to do as part
19 of your aging management program?

20 Is it the timing or is it the actual
21 nature of the inspections?

22 MR. CROUCH: The type of the inspections
23 will be the same. They'll be visuals or surface exams
24 or ultrasounds, whatever is appropriate for that piece
25 of equipment. The only real difference is that there

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1 will be a slightly larger scope and it will be focused
2 solely on the non-replaced equipment.

3 MEMBER POWERS: Thank you.

4 MR. CROUCH: This periodic inspection
5 program, it is one of the open items that we will talk
6 about here in a few minutes and the reason it is an
7 open item is this program has not been fully developed
8 as far as the exact scope and breadth of this program,
9 but the overall concept, everybody agrees on it and
10 we're just in the process of discussing with the staff
11 exactly where we're going to inspect, how often and
12 where.

13 So moving on to page 14. As we said, Unit
14 1 was shut down back in 1985 and placed in lay-up
15 status. There were systems that were placed in the
16 dry lay-up and systems that were placed in the wet
17 lay-up. The dry lay-up systems were configured such
18 that the systems were opened up and we blew
19 dehumidified air through the system to make sure that
20 the humidity in the system was low. We monitored the
21 humidity on the downstream end of where we were
22 blowing through. We also went through and monitored
23 the low point drains to ensure that there was no
24 standing water in the systems.

25 MEMBER POWERS: Your lay-up program

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1 started before the EPRI report was available. Was
2 your program consistent with the EPRI report after it
3 came out?

4 MR. CROUCH: Bob Moll, did we ever make
5 any changes as a result of the EPRI document coming
6 out?

7 MR. MOLL: No.

8 MR. CROUCH: Bob acknowledged that there
9 were no changes required.

10 We also had systems that were in wet lay-
11 up. These were primarily systems such as reactor
12 vessel where we maintained them full of water. In all
13 of these cases, we maintained the water chemistry in
14 accordance with the plant technical specifications, so
15 that the systems would have been experiencing the same
16 physical condition as if they had been in operation
17 from a chemical standpoint.

18 Many of the systems that were in wet lay-
19 up such as the recirc. piping, RWCU piping, portions
20 of the RHR and core spray piping that were out to the
21 isolation valves, all this piping has been replaced
22 anyway. The only major system or only major component
23 that was in wet lay-up that will still be present in
24 the plant will be the reactor vessel itself and that
25 component is obviously receiving large amount of

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1 scrutiny through the BWRBIP program. So we're
2 confident we know the conditions down in the vessel.

3 MEMBER BONACA: One thing that the ACRS
4 makes clear is that the early phase of the shutdown,
5 the lay-up wasn't as controlled as discussed here.
6 There is various inspection reports from 1987 talking
7 about inadequate lay-ups.

8 Do you have any comment on the impact of
9 those -- on that inadequate lay-up? I mean is it only
10 for components which have been replaced or doesn't
11 sound that way from the SER.

12 MR. CROUCH: The fact that we had
13 inadequate lay-ups was recognized and corrected by
14 making the lay-up processes in accordance with the
15 EPRI document. The systems that were affecting this,
16 they will be inspected, as we've talked about, so that
17 we can ensure that if there was any adverse effect
18 from the inadequate lay-up, we'll know about it and
19 respond appropriately.

20 MEMBER BONACA: That's important because
21 I mean that's one of the reasons why we're talking
22 about the periodic problem. I mean simply there was
23 a phase in which it's not understood whether you had
24 some latent effects that could have negative results
25 when you start operation of power.

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1 MR. CROUCH: Now in addition to systems
2 being in dry lay-up and wet lay-up, we had some
3 systems that were simply drained of water and they
4 were left at basic atmospheric conditions. We found
5 some instances where in two cases systems left in that
6 configuration did experience adverse conditions, in
7 particular, the system called the residual heat
8 removal service water system, which is a raw water
9 system that takes water from the river and is the
10 cooling side of the RHR heat exchangers. During Unit
11 3 recovery, we found that that piping inside the
12 reactor building was extremely degraded due to the
13 fact that it had moisture laden air inside it in a
14 basically warm environment. That piping required
15 complete replacement in Unit 3.

16 When we went over to do Unit 1 recovery,
17 we experienced the same mechanism. We knew it was
18 there before we even started Unit 1 recovery and so
19 all the affected piping in Unit 1 like that has also
20 been replaced.

21 We also found instances where raw cooling
22 water piping that had been drained experienced
23 degradation because some of the isolation valves
24 allowed water to leak back into the system and you got
25 basically the same condition where you had a small

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1 amount of water in basins and airfield system in a
2 warm environment. And it was corroded to the point
3 that it was usable, so we're replacing approximately
4 3,000 feet of small bore rock cooling water piping
5 despite that.

6 We found that piping that was in-service
7 full water did not experience this severe degradation
8 because of the chemical treatment in the biocides that
9 were in the piping, so systems such as raw -- the
10 large raw cooling water piping that was still in
11 service, it was fine. It was just the smaller, small
12 portion that was taken out of service going to
13 specific pieces of equipment that were affected.

14 So we took the lessons learned from when
15 we laid up Unit 3 and applied into the Unit 1 recovery
16 to ensure that we had the full scope. And what that
17 did for us was and we'll talk about this more when we
18 get the operating experience, it ensured that we have
19 the full scope of systems that are required to be
20 maintained and replaced as far as Unit 1 recovery, so
21 the systems will be in good operating condition for
22 when we start the unit back up.

23 Moving on to page 15, there's some
24 examples there are of various systems that were in the
25 dry and wet lay-up condition, but what I want to draw

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1 your attention to on this slide is the very last
2 bullet down there. As it says "no credit was taken
3 for the lay-up program in determining acceptability
4 structure systems and components for Unit 1 restart."

5 As we were talking yesterday, a better way
6 of saying this is that the lay-up program is the sole
7 basis for being -- for us saying that a system is good
8 for restart. In addition to having performed the lay-
9 up, we also are doing these inspections that we talked
10 about and we'll also be doing system testing as we
11 start up to ensure that the systems are capable of
12 performing their design functions.

13 What we mean by this bullet is we have not
14 used the lay-up as the sole basis for making sure a
15 system is good. We will demonstrate that it's good
16 either through visual inspections or system testing.

17 Moving on to page 17. As we talked about
18 Unit 1, 2 and 3 were shut down back in 1985. Unit 2
19 was recovered in 1991. Unit 3 was recovered in 1995.
20 And then they have operated since that point in time.
21 So Unit 1 has approximately 23 years of actual
22 operating, calendar years -- we're up to 22 years.
23 Unit 2 has approximately 23 years of actual operating
24 experience. Unit 3 has 18 years of actual operating
25 experience. That's calendar years.

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1 Unit 3 also experienced approximately 10
2 years of shutdown and lay-up, lay-up in the very same
3 kind of conditions as what we've seen for Unit 1. So
4 having experienced and extended shutdown in Unit 3
5 for 10 years, we were able to see basically whatever
6 type of lay-up effects that you would see, shutdown
7 effects, would have matured to the point that they
8 would stabilize before we started the unit back up.
9 So we're confident that the information that we gained
10 by recovering Unit 3 is directly applicable to Unit 1.

11 As Unit 3 started back up, and has now run
12 for 10 more years, after its long period of shutdown,
13 we have seen no unexpected effects of the layout that
14 the units have come up and they've performed very
15 well. We have seen no unusual degradation that we can
16 attribute directly back to the lay-up.

17 As we talked about the lay-up experience
18 from Unit 3 has been incorporated in Unit 1, talking
19 about the RHR service water and the small bore piping.

20 When we get ready to restart Unit 1, the
21 licensing basis for Unit 1 will be the same as what we
22 have in Units 2 and 3. As we talked about back on
23 that Appendix Foxtrot where we've got the 13 programs,
24 things that bring Unit 1 back into conformance with
25 Unit 2 and 3. Therefore, the operating experience

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1 that we have on 2 and 3, actual operation as well as
2 the shutdown, lay-up and restart experience, will be
3 directly applicable to Unit 1. So we're confident
4 that even though Unit 1 does not have the legally
5 required 20 years of operating experience, we have
6 operating experience from sister units that will tell
7 us the condition of Unit 1.

8 Coupling that with the periodic Unit 1
9 inspections we talked about, we're confident we will
10 know the condition of Unit 1 and be able to detect any
11 unexpected aging effects as we go through there. As
12 we restart Unit 1, its overall design, configuration,
13 operating procedures, text specs, FSAR and everything
14 will be identical to Units 2 and 3.

15 So once we get done, Unit 1 will be
16 accumulating its own operating experience under the
17 same operational conditions as what Unit 2 will be
18 experiencing.

19 Moving on to page 19. Through the course
20 of the license renewal process, we made various
21 commitments to approximately 114 commitments made
22 today and these are being tracked in both our on-site
23 commitment tracking system and our Appendix B problem
24 identification, problem evaluation report or
25 corrective action program systems. This will ensure

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1 that the commitments we made for program
2 implementation will get implemented as we permitted
3 it.

4 Moving on to page 20, as Dr. Kuo talked
5 about, there are four open items currently. One of
6 these has to do with core plate hold-down bolts.
7 These are the bolts that hold the core plate down and
8 keep it from moving in the event of accidents and
9 transients. We're currently in discussions with the
10 staff about the analyses that were done to demonstrate
11 the fact that these bolts will be able to maintain
12 their strength in pre-load, following an extended
13 period of operation.

14 The second one has to do with the drywell
15 shell corrosion. What this deals with is up at the
16 top of the drywell, there is a set of metal bellows
17 that separates the refueling cavity from the drywell
18 down below. And the bellows keeps the water from the
19 refueling cavity from going down and getting on the
20 outside of the drywell shell.

21 The staff has requested that we conduct
22 additional inspections of the shell. We have already
23 made inspections in the past and we feel that our IBE
24 metal containment program is sufficient, but we're
25 still in discussions with the staff to resolve this

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1 technical issue.

2 The third item has to do with the
3 inspection of some piping that's out in the intake
4 structure. The RHR service water piping out in the
5 intake structure is embedded piping and there was some
6 discussions between us and the staff as to how this
7 piping would be contained within an aging management
8 program. We originally made a statement that we would
9 inspect the pipe. We later realized the pipe was
10 embedded and could not be inspected from the exterior
11 and we were planning on doing an exterior inspection.
12 The staff desires that we do an interior inspection.

13 However, this piping is under our General
14 Letter 89-13 program and that it receives all of the
15 chemical injections for corrosion inhibitors for
16 microbiological inhibitors and the whole program is in
17 conformance with 89-13 which is what is required by
18 the Generic Aging Lessons Learned document. So we're
19 still in discussion with them as to the exact scope of
20 this piping.

21 MEMBER DENNING: With regard to that
22 piping, how does it relate to the piping that was
23 replaced in the HRR service water?

24 MR. CROUCH: This piping has been under
25 water the whole time. It is actually made so that the

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1 chemical injections to it happen immediately upstream
2 of that piping. So it has absolutely the highest
3 concentration of corrosion inhibitors and biocides at
4 that point in the whole system.

5 We also have coupons back in the system
6 that we can pull occasionally to monitor the condition
7 of the piping. We've been pulling those coupons and
8 they're not showing any evidence of corrosion or
9 microfiling either. So we are confident that the
10 piping itself is in good condition. But that's still
11 under discussion with the Region staff.

12 The fourth open item is this Unit 1
13 inspection program we've talked about. We're going to
14 consider that an open item to ensure that the program
15 gets fully scoped out and the details are in it so we
16 ensure that the plant is inspected properly.

17 So overall, we think we've put together a
18 program that is consistent between Units 1, 2 and 3.
19 We are confident that the program is consistent with
20 the Generic Aging Lessons Learned document. The
21 Appendix Foxtrot in the license renewal application
22 will ensure that Unit 1 will be operationally
23 identical to Units 2 and 3 from the standpoint of the
24 operators' concern as well as for license renewal.

25 We've taken the operating experience from

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1 Units 2 and 3 from both its operation and shutdown and
2 applied it to Unit 1 as part of the recovery as part
3 of the on-going operation to ensure that we know what
4 this plant's condition is going to be in the ensuing
5 years.

6 So any other questions?

7 MEMBER RANSOM: I have another question.
8 I just wanted to go through then this issue of the
9 transfer of operating experience and how it is -- what
10 additional actions are being taken to ensure that Unit
11 1 really has an appropriate either level of operating
12 experience or compensatory measures. The logic of the
13 initial inspection program only relates to nonreplaced
14 equipment, right?

15 MR. CROUCH: That's correct.

16 MEMBER RANSOM: And so in a sense, it
17 accounts for the possibility that during the period of
18 lay-up that there could have been things that were --
19 some mechanisms could have been initiated that perhaps
20 would show up --

21 MR. CROUCH: Once the system is turned
22 back in operation.

23 MEMBER RANSOM: Once the system is turned
24 back in operation. Now there's a lot of equipment,
25 however, that has been replaced and so on that

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1 equipment there is no operating experience, but is the
2 logic that much of that equipment has already been
3 replaced on Units 2 and 3 and so that the parallel
4 operation of 2 and 3 is provided, is that the logic?

5 MR. CROUCH: That is correct. The same
6 equipment that will be installed on Units 2 and 3,
7 using the same materials, so we have introduced no new
8 materials into Unit 1, not already present in Units 2
9 and 3.

10 MEMBER RANSOM: But there's no extended
11 period of operation yet of that new equipment?

12 MR. CROUCH: That is correct.

13 MEMBER RANSOM: So --

14 MR. CROUCH: One other program we didn't
15 talk about is there are periodic inspections for Units
16 1, 2 and 3. It's one of these other aging management
17 programs that will pick up that type of a situation.

18 If there are no further questions --

19 MEMBER POWERS: I've got another question
20 about the periodic inspections that you plan for Unit
21 1. If you do wind up operating the plant, why those
22 are going to be a mixture of current operating bases
23 as well as power uprated conditions.

24 MR. CROUCH: That is correct.

25 MEMBER POWERS: Would it be impossible to

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1 separate -- well, count that, I guess as experience,
2 I would think.

3 MR. CROUCH: Obviously, we will have to
4 look at the results and determine which of these are
5 where due to just operation as well as which of this
6 is aging effects and so that's the reason we've got
7 our various engineers, metallurgists, etcetera that
8 will look at these results to determine what is the
9 mechanism that's occurring here. Where -- like fact,
10 just purely due to the steam or is this some type of
11 a corrosion mechanism due to aging.

12 If there are no other questions, I would
13 like to thank you for the opportunity to come and talk
14 to you.

15 MEMBER BONACA: I think now we'll hear
16 from the staff, someone from the SER.

17 MR. SUBBARATNAM: Good morning. My name
18 is Ram Subbaratnam. And I am a project manager for
19 the Browns Ferry license renewal application. I'm
20 assisted by Yoira Diaz Sanabria and she'll be
21 presenting her portion of the open items. I'm also
22 assisted by Caudle Julian from Region 2, who helped us
23 with the AMR inspection.

24 Next slide, please.

25 (Slide change.)

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1 MR. SUBBARATNAM: Bill already explained
2 that we started out with the two open items in the old
3 SER based on the region's inspection in regard to one
4 more item added on, the out-of-service water piping
5 open item and then based on the discussion with the
6 subcommittee, we also added on the unit one periodic
7 inspection requirement as an open item.

8 As directed by the Committee, this
9 presentation is only related to the safety-related
10 matters of the license renewal application. As
11 previously stated, this license renewal request is of
12 the current uprate power level and does not include
13 external power uprate. This is only the fundamental
14 principles on which we based this evaluation.

15 The other principle is restoring the
16 current licensing basis of Unit 1 after completion of
17 those 13 Appendix F items. As long as these two
18 items, as I described, we met the fundamental
19 requirement to grant the license at the current power
20 level.

21 As suggested by the Committee yesterday,
22 we will only talk about the open items which remain on
23 our plate today.

24 Section 2.4-3, the drywell shell
25 corrosion, one of the items, TVA did a good job of

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1 explaining the mechanism of how the potential of the
2 postulated corrosion could occur. So I'm not going
3 into the details of that. What all we are trying to
4 do is the two staff options of what will be done.

5 One is to include the refueling cavity
6 seal in the scope of the license renewal so that this
7 will assure that the potential degradation of an
8 accessible side of the drywell is monitored and
9 managed. Alternatively, the staff would also like to
10 retain an option to periodically monitor the
11 degradation, if any, of the inaccessible side of the
12 drywell by using suitable testing matters like
13 ultrasonic testing. We are still in negotiation and
14 discussion with the licensee and we will find a
15 solution to this one.

16 VICE CHAIRMAN SHACK: I can understand
17 what I get from Option 2 where I monitor the
18 degradation. What do I really get from Option 1?

19 MR. SUBBARATNAM: Well, Option 1 also is
20 the same thing in a sense. The thing is the refueling
21 cavity seal currently for definition are not within
22 the scope of license renewal. So we are kind of
23 asking the licensee because of the operational issues
24 and problems with the potential degradation, we are
25 kind of going out to ask them to include it into the

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1 scope so it will include -- they will have to look at
2 it during every refueling outage to see how the
3 refueling seals are holding. There are 15 of them
4 which prevent the leakage going down to the sand
5 pocket area.

6 VICE CHAIRMAN SHACK: But does that mean
7 it will have to have some program that detects leakage
8 through the seals?

9 MR. SUBBARATNAM: We have to ensure
10 inspection of those things to see that what is the
11 condition of those leaks. If there is any water
12 accumulation in the sand pocket area, based on what
13 you see down the liner and go from there. And if we
14 think that the seal is bad or it's leaking, we
15 probably will ask them to do a corrective action
16 through the plant corrective action procedure to ask
17 them to replace those seals.

18 MR. KUO: Ram, I think Dave may want to
19 supplement that.

20 MR. JENG: I am David Jeng. The Option 1,
21 you raised the question if it was to be included in
22 the scope. That would kick in that this would be
23 covered in the current monitoring program, inspection
24 of the seals, so that would take care of that.

25 MEMBER POWERS: So you understand Option

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1 2, I guess I don't.

2 MR. JENG: Sir, could you repeat the
3 question?

4 MEMBER POWERS: I didn't ask it.

5 (Laughter.)

6 MEMBER POWERS: I guess the question is
7 what do we mean by periodically monitor the potential
8 degradation of the unit's inaccessible side.

9 MR. JENG: That's second option.

10 MEMBER POWERS: Let me be clear. I don't
11 understand what potential means in the sentence and I
12 don't understand inaccessible.

13 I mean how do you monitor something that's
14 inaccessible.

15 MR. JENG: Through the volumetric
16 inspection.

17 MEMBER POWERS: Again, I don't understand
18 -- if it's inaccessible, you cannot monitor it. That
19 would be definition of inaccessible.

20 MR. JENG: Well, in theory, there will be
21 action from inside the dry well, along with volumetric
22 inspection. It will tell me whether there is a
23 ceiling or not.

24 MEMBER POWERS: You plan to monitor the
25 actual degradation and not the potential degradation?

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1 MR. CROUCH: Let me see if I can help.
2 This is Bill Crouch, the site licensing manager from
3 Browns Ferry.

4 MEMBER POWERS: Maybe you can help. I'm
5 not getting help otherwise.

6 MR. CROUCH: The Browns Ferry containment
7 is a steel structure with concrete liner.

8 MEMBER POWERS: Son of a gun. Unusual
9 among BWRs, I take it.

10 MR. CROUCH: And so since it's got
11 concrete on the outside, you can't get to the outside
12 of the steel shell, obviously, but what's proposed to
13 do is to ultrasonically shoot through it to see if
14 there's any thickness degradation of the shell. So if
15 you shoot it from the inside towards the outside.

16 MEMBER POWERS: I guess I understood that.
17 Now how does that get to the potential degradation.

18 MR. CROUCH: I think it's the way they're
19 wording it. They would monitor degradation --

20 MR. SUBBARATNAM: That is probably
21 correct. Actually, they did have some experience of
22 what recommendation there was.

23 MR. JENG: When we say potential we mean
24 if there were no leaking of those seals, then of
25 course, there will be no degradation. That's why --

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1 MEMBER POWERS: Not monitoring seals here,
2 surely. Actually, we're looking at the thinning of
3 the steel. And it's just wastage that you will
4 detect, right? That raises another question.

5 Seems that the Taurus, the Fitzpatrick
6 didn't reflect any wastage, did it?

7 MR. JENG: This is drywell we're talking
8 about.

9 MEMBER POWERS: I understand that, but
10 steel is kind of steel, right?

11 MR. JENG: Yes, we have both kinds of
12 steel.

13 MEMBER POWERS: So is your ultrasound
14 going to work on cracks?

15 MR. JENG: Well, we are talking about the
16 material basis of the shell thickness.

17 MEMBER POWERS: So only on wastage
18 matters. It doesn't matter if this thing cracks?

19 MR. JENG: It's loss of the material
20 concern.

21 MEMBER POWERS: I mean, I take it your
22 answer means that cracks don't count; the only thing
23 that counts is wastage.

24 MR. JENG: There should be an environment
25 and conditions which would be conducive to such a

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1 situation.

2 MEMBER POWERS: You don't know the
3 conditions on that, so -- it's inaccessible. I'm
4 really confused. It's inaccessible, so you don't know
5 the conditions.

6 MR. JENG: We know the general environment
7 there.

8 MEMBER POWERS: Oh, can you give me all
9 the data you have on the general environment? Things
10 like pH, water content, chemistry of the water?
11 Include PHI and concentration, conductivity?

12 MR. JENG: The water aspect is controlled
13 by the water chemistry program, I believe.

14 MEMBER POWERS: I don't think there's any
15 water chemistry on the backside of that drywell.

16 MR. JENG: No, this water come from the
17 reactor, you know, during the refueling operations.

18 MEMBER POWERS: Dr. Shack, I think you
19 understand my confusion on this second option?

20 VICE CHAIRMAN SHACK: The assumption is
21 that the degradation is wastage rather than fraction.
22 There's no mechanism for fatigue here, really. There
23 is a possibility of stress corrosion cracking, but
24 that does seem unlikely in a carbon steel in this kind
25 of environment so that certainly the most like

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1 mechanism is wastage which is what they're really
2 protecting.

3 MEMBER POWERS: I'm struggling to
4 understand how they understand the environment, since
5 it's inaccessible.

6 MR. JENG: Inside air environment, inside
7 containment air environment.

8 MEMBER POWERS: There are lots of
9 varieties of air in this world.

10 MR. SUBBARATNAM: We will take this
11 question under advisement. Before we come back we'll
12 have hopefully a better answer for you, sir.

13 MEMBER POWERS: Thank you.

14 MR. SUBBARATNAM: The other open item is
15 Section 3.7 on the periodic inspection. Bill Crouch
16 explained in detail how the evaluation came about. He
17 did explain that we met or exceeded the EPRI
18 requirement for that.

19 I will briefly describe why did staff and
20 the facilitator exceed beyond the EPRI content. They
21 said the staff needed additional information from the
22 applicant to conclude that no new degradation occurred
23 in the external outage. Specifically, the staff
24 requested the following information, that most severe
25 aging did not occur during the extended outage. And

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1 two, additional agings are properly identified,
2 evaluated and managed and proposed the -- proposed
3 management can distinguish the aging during the
4 extended period from the aging during the future
5 operation.

6 These are the three basic questions with
7 which staff explored the SER and that led to the Unit
8 1 periodic inspection. We are still in dialogue with
9 the applicant in finalizing a few of details and what
10 staff is briefly looking at only is the scope of the
11 program, the sampling basis, the aging effects and of
12 course, monitoring and trending.

13 Bill Couch very briefly said that we going
14 to have three occasions when we are going to look at
15 and do a monitoring and trending. We will have
16 finalized details when we come back to the Committee
17 again. And then also, an operating experience
18 commitment.

19 So these are all the five items we need to
20 finalize before we can finalize the details of this
21 program.

22 MEMBER BONACA: Can you go to the previous
23 slide? It says, "BFN submitted Unit 1 periodic
24 inspection program."

25 MR. SUBBARATNAM: They actually were --

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1 that is true, Dr. Bonaca. What you don't see in the
2 safety evaluation, this happened after we wrote that
3 item.

4 MEMBER BONACA: I understand. We are
5 commenting on the document interaction and there's a
6 statement there that's not identified as a commitment
7 yet, all this stuff has to happen in the SER.

8 MR. SUBBARATNAM: Right. And when we plan
9 the SER, you will have a new program evaluation which
10 will be the 39th program. We have evaluation for 38
11 so far.

12 MEMBER BONACA: We'll look at it then.

13 MR. SUBBARATNAM: Yes.

14 MEMBER BONACA: Okay.

15 MS. DIAZ SANABRIA: Good morning, I'm
16 Yoira Diaz Sanabria. I'll be discussing the open item
17 in stress relaxation core plate hold-down bolts.

18 The evolution of the issue started when
19 the staff requested additional information of the
20 applicability of BWRVIP-25 loss of preload criteria
21 for the core plate hold-down bolts due to thermal and
22 irradiated effects.

23 In its response, the applicant has
24 specified that the analysis was evaluated at the
25 assumed expected loss of preload of 20 percent, which

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1 bounced the original BWRVIP-25 value. The applicant
2 indicated that core plate hold-down bolts will
3 maintain sufficient preload to prevent the sliding of
4 the core plate by friction under normal and accident
5 conditions. The bolts also met their ASME, Section 3,
6 Class 1, level D limit at the end of the period of
7 extended operation.

8 After the staff review, the method of
9 analysis, based on GE's plan specific stress
10 relaxation analysis on irradiated stainless steel
11 materials, requested additional information for the
12 following: horizontal and vertical loads for all
13 operating conditions, prevention of the sliding of
14 core plate due to friction and in our handouts, you
15 have a different second bullet, which we modified
16 yesterday after talking with the staff. It really is
17 the prevention of the sliding of core plate due to
18 friction. And the third one is axial and bending
19 stresses.

20 The staff have not yet received the
21 information about the mention of Applicant's steel
22 ongoing on its review since this is proprietary
23 information coming from GE's open request. Then this
24 issue is still open and we're waiting for the response
25 from the applicant. This is our understanding.

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1 Any questions about the open item?

2 MR. SUBBARATNAM: If there are none, we'll
3 turn now to Caudle Julian and he will describe some of
4 his AMR instruction details for us.

5 MR. JULIAN: Thank you. My name is Caudle
6 Julian from NRC Region 2 and I was the time leader for
7 the Aging Management Program Inspections, License
8 Renewal Inspections at Browns Ferry.

9 The first inspection we did at Browns
10 Ferry was conducted November 29th through December
11 17th and the inspection concluded overall that the
12 existing programs which they're going to credit as
13 aging management programs, were indeed functioning
14 well. The inspectors observed that the applicant had
15 not yet begun the implementation process for the new
16 AMPs, aging management programs, in that AMP
17 procedures had yet to be defined and proposed and for
18 the existing programs identification and selection of
19 which particular existing procedures constitute the
20 AMP had yet to be done. Region 2 concluded that the
21 NRC needed to perform another inspection at Browns
22 Ferry.

23 We did do a good bit of walking down the
24 plant systems during that visit and in walkdowns of
25 the plant systems, we concluded that the plant

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1 equipment is being maintained adequately.

2 MEMBER KRESS: On your second bullet
3 there, what does "generally" mean. Does that mean
4 there are particular ones that are not functioning
5 well?

6 MR. JULIAN: Oh, generally functioning
7 well means as we would see in the norm, in the
8 industry, equivalent to other plants. We go in and we
9 inspect

10 --

11 MEMBER KRESS: There's no distinction
12 between general and specific then?

13 MR. JULIAN: No.

14 MEMBER KRESS: Okay.

15 MR. JULIAN: If we go in and start looking
16 at ISI programs, fire protection, etcetera, etcetera
17 and sampling things out of there, we're going to
18 detect some little flaws here and there in
19 documentation or performance at any plant. But we
20 thought that Browns Ferry's was on a par with other
21 people that you looked at.

22 MEMBER KRESS: Okay, thank you.

23 MR. JULIAN: So we thought the material
24 condition was being maintained at Browns Ferry
25 adequately.

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1 We went back to the second inspection,
2 September 19th to 23rd, let's go to the next slide,
3 please.

4 (Slide change.)

5 MR. JULIAN: There you go. We looked at
6 a sample of aging management programs. I counted 40.
7 I was just looking off an old list, I guess. They say
8 there's 39. They had put together implementation
9 packages for each of the aging management programs.
10 The packages, we found, contained some errors and we
11 concluded that they were not meticulously reviewed.

12 The applicant initiated a PER, that's a
13 corrective action document, a condition report,
14 essentially, other words are used at other plants, to
15 address this under their Corrective Action Program and
16 go back and look at the scope of the problem, since we
17 weren't working on the sampling basis.

18 Next slide, please.

19 (Slide change.)

20 MR. JULIAN: We looked at their plans for
21 tracking future actions using their TROI system.
22 That's the system that TVA has had for years and years
23 and years which is an electronic system for capturing
24 action items that are mainly coming out of licensing
25 activities.

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1 The system and when we got there was not
2 initially linked to the implementation packages, that
3 is, the file record number for the implementation
4 package did not appear in the action items so it would
5 be hard for a person years hence to go back and track
6 exactly what did they want us to do. When we pointed
7 that out, they quickly corrected that within a day.

8 The inspection sample that we selected --
9 VICE CHAIRMAN SHACK: Was that unique to
10 this or is that a feature of their tracking system?

11 MR. JULIAN: I don't think that it was
12 unique to this particularly. It's a free format that
13 they have in their tracking system. It depends on the
14 author to put down what he thinks is necessary to --

15 VICE CHAIRMAN SHACK: How do you track
16 without making that link?

17 MR. JULIAN: It looks to us like these
18 items were added probably into TROI as they were going
19 along through the review process and there were no
20 implementation packages and then they really turned to
21 and built these implementation packages, but have not
22 yet got around to going back and putting the
23 references into the system.

24 We took our sample of inspection
25 commitments and looked through the stack of paper

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1 about that thick of TROI items and we were able to
2 find a tracking method for everything that we sampled,
3 but it was hard. There was much duplication in the
4 system. There was varying formats and loading items
5 in. For example, one place there would be three
6 separate items for Unit 1, Unit 2, Unit 3. Another
7 place, there would be one item for do this on all
8 three units. And it was not very user-friendly in
9 that you're doing random search just through a pile of
10 paper, trying to find the commitment that you're
11 after.

12 The applicant recognized that and decided
13 to track this, again under their formal corrective
14 action system and then writing a PER on it. And we
15 concluded we'd like to go back, Region 2 would like to
16 conduct another inspection to see the results of that
17 effort, to ensure that they have indeed everything
18 captured as best we can see.

19 One technical issue that came up during
20 that discussion, we talked about a lot and Bill Crouch
21 talked about earlier is the RHR service water piping.
22 We recognized during the first inspection that there
23 are the water that flows from the river into the
24 chamber which is the suction for the RHR service
25 water, all the safety-related pumps in the plant flows

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1 through three 24-inch diameter cast iron pipes, about
2 40 feet long, that are cast into the concrete of the
3 intake structure. We raised the question wouldn't it
4 be a prudent thing to do some sort of inspection on
5 those pipes since they apparently have never been
6 looked at and at first, in the first inspection we
7 thought we had agreement that TVA would do a one-time
8 inspection to look at those pipes to see that nothing
9 bad is going on, the pipes are not corroded away, so
10 we're gradually eroding away the concrete. There's
11 not been some sort of build up of material in there
12 that's choking those pipes down or anything else
13 that's not going on, it's an aging effect.

14 When we came back the second time, TVA had
15 decided that they did not want to do such an
16 inspection. They don't think it's necessary because
17 they don't think that these things can suffer bad
18 aging effects because of their design and it's too
19 hard. That's what it amounts to. We considered the
20 possibilities of divers doing it, but it's probably
21 too dangerous because we're in an environment where
22 they're operating pumps and other pumps that might
23 automatically start in a hurry.

24 We do not advocate putting people in
25 danger to do such inspections, but we think there are

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1 various ways using TV cameras and other remote
2 mechanisms that such an inspection could be done and
3 so right now TVA has written a PER on this item and is
4 considering how to resolve that issue or writing down
5 a technical discussion of why they think, don't think
6 that such an inspection is necessary.

7 We'll be working in the future with NRR to
8 help to resolve this issue. So that's something we'd
9 also like to look at when we go back to the next
10 inspection.

11 Next slide.

12 (Slide change.)

13 MR. JULIAN: The conclusion is that NRC
14 will perform another inspection when the applicant has
15 progressed further with AMP development
16 implementation. And in walking down plant systems and
17 examining plant equipment, the inspectors found no
18 significant adverse conditions. It appears that plant
19 equipment was being maintained adequately.

20 That concludes what I had to say. Are
21 there any questions?

22 MR. SUBBARATNAM: Dr. Bonaca, that
23 concludes the staff's presentation.

24 MEMBER BONACA: Any additional questions
25 from members or members of the public? If none, I'll

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1 turn it over to --

2 MEMBER POWERS: Let me ask just one area
3 that I'm being somewhat curious about. There are
4 bellows on the downcomers from the drywell into the
5 Taurus. Could those be inspected?

6 MR. SUBBARATNAM: Actually, Dr. Powers,
7 the staff who was dealing with the safety evaluation
8 on this particular aspect is not there today. David
9 was just filling in for him. I will go back to the
10 staff and ask them. He did mention there was an
11 inspection done on the bellows.

12 MEMBER POWERS: If you can give me the
13 outcome of that inspection, I'd appreciate that.

14 MR. SUBBARATNAM: We'll do that and take
15 it under advisement and we'll give you a current
16 answer for next time.

17 MEMBER POWERS: Appreciate it.

18 VICE CHAIRMAN SHACK: We're ahead of
19 schedule.

20 MEMBER KRESS: So we're going to write an
21 interim letter here, is that true?

22 VICE CHAIRMAN SHACK: We do plan to write
23 an interim letter in this case.

24 MEMBER KRESS: Okay.

25 VICE CHAIRMAN SHACK: And Mario has a

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1 draft that we'll be discussing later on today.

2 We're essentially on a break now until
3 10:15 if there are no further questions or discussion.

4 (Off the record.)

5 VICE CHAIRMAN SHACK: We're back into
6 session. We're now going to discuss the proposed
7 recommendations for resolving generic safety issue
8 GSI-80, "Pipe Break Effects on Control Rod Drive,
9 Hydraulic Lines and the Dry Wells of Boiling Water
10 Reactor Mark 1 and 2 Containments" and Jack Sieber
11 will lead us through this discussion.

12 MEMBER SIEBER: Okay, thank you, Mr.
13 Chairman. this issue has been around since 1978 and
14 was actually instigated by this Committee at that time
15 and it is one of a dwindling number of general issues
16 as the staff has been working them off.

17 This one is particularly interesting. The
18 concern with the smaller containments is just that.
19 The containments are small, the pressures go higher
20 and the heat absorption and the rejection capability
21 is challenged a little bit more than in the larger
22 containments. The issue here is if there is a LOCA
23 which impacts the hydraulic lines in boiling water
24 reactors, the hydraulic lines which control, provide
25 the motor power to the control rod drive mechanisms,

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1 the question is will the reactor scram or not, because
2 when you have a LOCA you would like the reactor to
3 shut down, particularly if you're going to inject cold
4 water to it, to cool the core. If you have excess
5 reactivity that will give you a cold water accident.
6 The core will overheat and you end up with a major
7 problem. So that's basically the issue.

8 Now for other control rod drive mechanisms
9 that work is that they have a high pressure heat line
10 and they have a discharge line and the reactor rods
11 will actually be inserted, even if the high pressure
12 line is broken because it can use the pressure inside
13 the reactor vessel to operate the control rod. And if
14 you break the discharge line, that's okay too, because
15 the water will just dump out on the floor and the rod
16 will still insert. The big problem comes is if you
17 somehow block the discharge line so that it can't
18 discharge the water, then the rod won't insert into
19 the core and the resolution of this issue looks at the
20 various ways and the probabilities of either crimping
21 the line shot or otherwise preventing the water from
22 coming out of the discharge line.

23 So with that introduction, that's
24 basically what the crux of the problem involved in
25 generic safety issue 80 is. I would like to introduce

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1 Jack Rosenthal from the staff to give us further
2 introduction.

3 MR. ROSENTHAL: Jack Rosenthal, Advance
4 Reactor and Regulatory Effectiveness Branch in the
5 Office of Research.

6 Given your introduction, I really don't
7 know that I have very much more to say. I just wanted
8 to call your attention to the fact that the decision
9 lingered with us for a while, with the vulnerability
10 identified and just recently Abdul Sheikh did some
11 ANSYS calcs which you'll hear about and that provided
12 a real engineering implement -- increment -- that
13 allows us to resolve the issue. That's a big change
14 that's happened in the last year.

15 With that, Harold?

16 MR. VANDERMOLEN: Thank you, Jack. My
17 name is Harold Vandermolen. I work for the Generic
18 Issues Program. On my left is Mr. Abdul Sheikh, who
19 works with the Division of Engineering Technology and
20 yes, we've -- it is indeed a very interesting issue.
21 I'd like to start out by reviewing a little bit with
22 the sequence of events that happens in this --

23 MEMBER APOSTOLAKIS: Is it clear to
24 everyone why this has been a safety issue for so many
25 years? You said 1977?

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1 MEMBER SIEBER: It was rated as a low
2 priority for many years.

3 MR. VANDERMOLEN: I'm going to touch on
4 the history in just a moment.

5 MEMBER APOSTOLAKIS: You will?

6 MR. VANDERMOLEN: Yes.

7 MEMBER APOSTOLAKIS: Okay, thank you.

8 MR. VANDERMOLEN: In the course of the
9 scenario under consideration, we start out with a
10 classic large break LOCA in a boiling water reactor
11 and in this particular scenario, it was noticed that
12 some of these pipes come very near some of the control
13 rod drive hydraulic lines. Now a boiler will have
14 something on the order of 180 control rods, so it's
15 quite a nest of these lines. The hydraulic control
16 units are located outside of primary containment and
17 each one has to be connected to its control rod drive
18 through two lines. So in certain areas around the
19 vessel support skirt you have quite a bunch of these
20 going in.

21 Now again, it's true, breaking the lines -
22 - it was designed as such that breaking these lines is
23 not going to be a problem. But crimping them shut
24 could give you a problem and what we worry about is
25 the ECCS system that in refilling the reactor with

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1 cold water, and some of the rods being left behind.
2 The way the issue was first posited, we were going to
3 -- we were supposed to worry first about a possible
4 reactivity excursion, which turns out to be not that
5 much of a problem, but also the fact that it's an
6 additional post-LOCAL heat source which is potentially
7 more of a problem. We'll be going into this in
8 considerably more detail in just a moment.

9 If we can go on to the next slide, I'll
10 just give you a heads up.

11 (Slide change.)

12 MR. VANDERMOLEN: After a lot of work,
13 particularly by Mr. Sheikh next to me here, we did
14 discover that the core damage frequency was well below
15 the thresholds and the public risk was also well below
16 the thresholds for us to actually take regulatory
17 action.

18 I'd like to review the history of this
19 issue a little bit. It was actually raised formally
20 as an inherent issue by the ACRS in 1983 and actually
21 was first discovered earlier than that when an ACRS
22 member on a plant tour noticed that some of the large
23 break -- large pipes were rather close to the next of
24 control rod drive hydraulic lines and started asking
25 well, can you really picture these things remaining

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1 operable if this huge pipe right next to it with 1,000
2 psi inside of it has burst? Very good question.

3 We did look at it, beginning of 1983, and
4 we did a -- what we now call a screening analysis.
5 Back then we called it prioritization. It was a
6 calculation that was reminiscent of the sort of thing
7 you should do with nuclear cross section work. As we
8 look at the sort of target area of the nest of control
9 rod drive hydraulic lines and how much -- it's not
10 really a solid angle because you don't go in four or
11 five directions, but how much of an area is subtended
12 from potential sources of -- where the pipe would
13 break and we were worried about things like pieces
14 breaking off, missiles, and things like that.

15 Purely on these semi-geometrical
16 arrangements, we got a fairly low core damage
17 frequency and we prioritized it as low priority in
18 1984. Now this is one -- at that time approximately
19 400 generic issues, so it didn't get to the top of the
20 priority list for many years. What happened then in
21 1995 it was closed out. That is not because we got
22 tired of it. Nor did we change anything in our
23 analysis. What happened was in 1995, the agency had
24 a policy change where we switched from valuing a
25 person rem at \$1,000 to \$2,000. And all of these --

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1 there was sort of a blank action. There were many
2 staff projects that were based on risk-informed
3 priority considerations and this then descended into
4 the draft category.

5 Well, it didn't stay there very long. In
6 1998, we had a team of people going out, actually
7 working on a different issue, Generic Issue 156-61 and
8 going through some plants looking at piping layouts
9 and they discovered some Mark 1 boilers that not only
10 had the pipe, the large recirc pipe near the control
11 rod drive hydraulic lines, but actually going through
12 the middle of the nest.

13 And the project manager for that generic
14 issue came to me and said did you know about this?
15 Was this covered in your original analysis? I said
16 well, no. I actually said a few other things, we
17 won't go into that, but it was a bit of a surprise and
18 immediately we said we've got to take a look at this
19 and fortunately we still had that team available and
20 so while they were doing that generic issue, we asked
21 them to start collecting some data for Generic Issue
22 80 as well.

23 And they did put together a NUREG in 1999,
24 identifying these breaks and actually reassessing the
25 priority. It's quite a conservative calculation, but

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1 we reopened the issue at that point.

2 So George, maybe I should stop. Have I
3 answered your question now?

4 MEMBER APOSTOLAKIS: Yes, that's good.
5 Thank you.

6 MR. VANDERMOLEN: So it has been a very --
7 whoops. It has been indeed an interesting process. It
8 was obvious we had to go through a completely
9 different approach. You could not simply write this
10 off based on geometry because it simply was not true.
11 So we went to the Division of Engineering Technology
12 and started asking them well, just what can happen if
13 you have an impact of this nature and at this point
14 I'm going to turn it over to Mr. Sheikh who is going
15 to describe some of these calculations.

16 MR. SHEIKH: Okay, so I started with this
17 and I looked -- the objective of the assessment was to
18 perform a detailed analysis to see what's the
19 interaction between the big RCS and RH piping with the
20 CRD pipe. And then look whether that after the
21 impact, whether the CRD piping can be crimped,
22 completely shut before it breaks or it will have some
23 space still there before it breaks. So that is the
24 key issue in this analysis.

25 And based on that, this was a

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1 deterministic approach. But then we went on further
2 and also developed a probabilistic approach to it to
3 determine the CDF, core damage frequencies, for this
4 piping. And then we compared the core damage
5 frequencies with the management directive 6.4
6 recommendations.

7 The RHR and RCS piping for Mark I and Mark
8 II -- inside the containment is essentially the same
9 as shown on the next page. There's no basic
10 difference in the routing of the piping inside the
11 containment or the drywell.

12 The differences are in the layout of the
13 CRD piping. The older plant Mark 1DII has three sets
14 of CRD bundles as shown on page 8 and they come out
15 from one side of the reactor. The other plants, next
16 page -- the other plants have four sets of bundles and
17 they come out diagonally off of it and in this picture
18 on the plant I've shown only two coming out, but they
19 are symmetric, two bundles one the other side.

20 So the way we did this assessment, we kept
21 the approach which was originally followed in the
22 NUREG 6395 which was issued in 1999 which identified
23 these issues as medium priority and high priority.

24 For calculating the core damage
25 frequencies, we have done some work which Harold has

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1 done which is based on the most conservative approach
2 which is the NUREG 1150 values, but just to have an
3 idea in the detailed assessment which we passed on, we
4 looked at other frequencies which have been developed
5 since then which are two orders of the magnitude than
6 the NUREG 1150.

7 The core damage frequencies is dependent
8 on four items which is initiating event and the next
9 item which they called and I followed the same names
10 as in the NUREG CR6395, the next one is PIPETYPE and
11 what they considered the PIPETYPE to be, the number of
12 pipe breaks in that RHR or RCS system as a fraction of
13 total number of five breaks in the high energy lines
14 inside the containment. So that's one factor.

15 The next factor is the TYPEFRAC and I have
16 -- we have these numbers in the detailed assessment
17 report. Then the next number is the TYPEFRAC which is
18 the fraction of RHR or RCS pipe that can impact on the
19 CRD piping. What we looked at, what represented the
20 plants for different GE models and we looked at the
21 total length of RCS or RHR piping and then looked at
22 where the breaks are predicted in the RCS or RHR
23 piping and looked at how much of that piping can
24 impact the CRD bundles and calculated that fraction.

25 And then the last item is the RUPTPROB

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1 which is the probability of this RCS piping or RHR
2 piping causing a complete blockage of the CRD piping.
3 That's the issue we have addressed before.

4 We were not worried that if they bend the
5 pipe or they have a smaller gap. Next page.

6 (Slide change.)

7 MR. SHEIKH: So we looked at the different
8 containments. The first one was a GE2 containment,
9 Mark I which is the oldest plant and looking at the
10 layout of these plants, because the CRD piping is
11 located on one side, the RHR piping was not --
12 couldn't impact the CRD bundles. The RCS piping could
13 impact the CRD bundles and this is shown -- I don't
14 have it, but it's in the assessment, but if you go
15 back to page 8, it's the pipe sits in between the two
16 top bundles. The picture is there in the assessment
17 --

18 MEMBER SIEBER: Is it shown?

19 MR. SHEIKH: The pipe sits somewhere. So
20 we looked at the possibility and it's more or less --
21 the layout is similar to the picture shown on page 14.
22 You can see the CRD bundles. This is the CRD bundles.

23 MEMBER SIEBER: You will have to talk into
24 the microphone.

25 MR. SHEIKH: So anyway, the CRD bundles

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1 are there and this is the pipe break. And the break
2 locations, as I said, it's about 18 feet from the CRD
3 bundles and there is a gap at this point of 25 inches.

4 VICE CHAIRMAN SHACK: When I looked at
5 that, I was trying to figure out why I didn't end up
6 with that final reflected shape. There's a sort of a
7 white cross up there at the top. That's not a stop of
8 any sort. This thing is just straightened out so the
9 jet force is not bending, the moments are balanced and
10 that's the equilibrium configuration of the pipe.

11 MR. SHEIKH: Yes. I don't know how this
12 white mark is --

13 VICE CHAIRMAN SHACK: Okay, it's just
14 there.

15 MR. SHEIKH: Right. This is another GE5
16 plant, but this is very similar to the GE2 plant. And
17 as you can see, the pipe is not going to hit the CRD
18 bundles, but we went a step further and we assumed it
19 hits the bundle. And we looked at -- we put a small
20 force, a very small force on the RCS pipe.

21 VICE CHAIRMAN SHACK: The jet force is 600
22 to a 1000 kip and you put a 1 kip force?

23 MR. SHEIKH: Right.

24 VICE CHAIRMAN SHACK: Why?

25 MR. SHEIKH: Because you can see, once the

1 -- the pipe is so flexible and any force is going to
2 bend it. As you can see on page 12. And the pipe was,
3 this pipe has deflected almost 90 degrees by a force
4 of a thousand pounds. So if I put more force, it's
5 going to break.

6 The idea is to show it as it bends, you
7 can still see that there's no complete blockage there
8 and this is also documented in the famous work done at
9 one of the national labs and passed on pictures which
10 shows that they did some actual test on pipe to pipe
11 impact. And they all show that the pipes never crimp
12 completely blocked.

13 VICE CHAIRMAN SHACK: But then my question
14 there was, okay, you demonstrated that for 3-inch pipe
15 and you recorded the result for a 4-inch pipe, but
16 isn't it easier to crimp a 1-inch pipe?

17 MR. SHEIKH: No, it's actually the
18 reverse.

19 VICE CHAIRMAN SHACK: It's harder?

20 MR. SHEIKH: Right. Because the stiffness
21 of the 1-inch pipe is in bending, is much smaller than
22 in the crushing, you know, as you can imagine, if you
23 have a smaller diameter and you're pushing it with a
24 bigger diameter, it's much harder to crush. Before it
25 crushes, it bends.

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1 VICE CHAIRMAN SHACK: But I envision it
2 bending so much that it's some plastic hinge that
3 forms the kink.

4 MR. SHEIKH: Yes, but before it forms a
5 hinge, it breaks. And --

6 VICE CHAIRMAN SHACK: Okay, I guess that's
7 -- is it clear that a 1-inch pipe will break faster
8 than a 4-inch pipe or a 3-inch pipe?

9 MR. SHEIKH: That is true, because the
10 stiffness -- I mean the ultimate capacity of the pipe
11 is dependent on the stiffness of the pipe and that
12 stiffness is based on the diameter. It's the diameter
13 to the top power of 4 is the stiffness.

14 VICE CHAIRMAN SHACK: Yes.

15 MR. SHEIKH: So a 4-inch pipe doesn't
16 break, I mean completely shuts. One-inch pipe cannot.
17 And this report is 5 to 5 impact, 6395. I have shared
18 it with you. They also did this ANSYS's work on it,
19 previously, and they came up with the same conclusion.

20 But let me find out. We are going
21 defense-in-depth. Number one, we have shown that
22 there is no realistic possibility of the pipe
23 impacting the bundle. Then we are saying if it
24 impacts, it's not going to crimp.

25 MEMBER DENNING: Can you take us back

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1 again to the Slide 14. There you are saying based
2 upon that, that the pipe isn't going to impact the
3 bundle. What if the pipe break had occurred at the
4 other end, is there a possibility of a break
5 configuration in which you would have forces moving
6 that pipe up?

7 MR. SHEIKH: Pipe breaks are based on our
8 Reg. Guide and they are for the RCS, the breaks are
9 identified with the stress levels and this is the
10 breaks we are considering is the break at the nozzles.

11 MEMBER DENNING: So you don't allow the
12 breaks to occur in places other than what you consider
13 to be the high stress levels like in a nozzle?

14 MR. SHEIKH: And you can't have a break up
15 in the center of the pipe. These have to be -- the
16 breaks have to be at the nozzle.

17 MEMBER DENNING: It's just not allowed,
18 huh?

19 (Laughter.)

20 I mean you physically don't think you can
21 break a pipe there is what you're saying?

22 MR. SHEIKH: And that is the basis of all
23 the OSEP 3 plants. We don't consider a break in the
24 middle of the pipe. This is not special for these.
25 We do the same thing for the plants which are licensed

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1 after this generic issue.

2 MEMBER DENNING: For any pipe crimp
3 analysis.

4 MR. SHEIKH: Right. Anyway, so after we
5 have done this analysis and we concluded that the pipe
6 will bend without significant crushing or crimping
7 before rupture and as was mentioned and passed out
8 these pictures, this behavior is consistent with what
9 we have observed in previous tests.

10 However, to get to the PR core damage
11 frequently, we came up with the arbitrary number of .1
12 for this factor RUPTROB to determine what Harold has
13 determined the value is.

14 MEMBER DENNING: Okay, so this is
15 equivalent to saying that a fracture somewhere in the
16 middle there is 10 percent of the probability that
17 you'll get a fracture at the nozzle, assume it's not
18 impossible to break the straight pipe.

19 MR. SHEIKH: Right. And these on top of
20 it, Harold's calculations are based on NUREG-1150 and
21 all the draft NUREGs which we are going to be
22 publishing soon. For large damage of pipes, the
23 probability of failure is two orders of magnitude
24 higher. So even if you consider a problem with 1, 2
25 is still okay.

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1 So let me just carry on with the next one.

2 (Slide change.)

3 MR. SHEIKH: This layout, is like we said,
4 is for the other newer plants with a GE3, GE4 and GE5
5 and essentially the two sets of bundles run parallel
6 on each side of the reactor and there is an RHR pipe
7 up there as shown in Section 8 and it can break.
8 There, again, the way the analysis is done is it's a
9 guillotine break on the RHR on these lines and when
10 you have a guillotine break, the pipe breaks straight
11 in the direction. It's assumed to break straight in
12 line of the pipe.

13 So as you can see in picture, there is
14 very little series of gaps, 12 to 15 inches here, so
15 if the break occurs here, the pipe is going to go
16 straight and the likelihood of this hitting this
17 bundle which are separated by 15 inches is at least in
18 the deterministic approach, we don't consider it.

19 MEMBER DENNING: But my impression looking
20 at pipe whips is they go all over the place. Am I
21 wrong?

22 VICE CHAIRMAN SHACK: Like a firehose.

23 MEMBER DENNING: Yes, like a firehose. Is
24 that not true?

25 MR. SHEIKH: These are postulated breaks.

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1 If you have a pipe break, we are assuming that it's a
2 certain guillotine break which means the whole
3 circumference breaks.

4 MEMBER DENNING: Yes, but don't you have
5 to do it in the most conservative manner in terms of
6 thinking of well, that there could be lateral --

7 VICE CHAIRMAN SHACK: A zone of influence
8 that looks like a cone?

9 MR. SHEIKH: If you see the guidelines the
10 way it is done it's straight. That's the force that's
11 taken, going straight in the axis of the pipe.

12 MEMBER APOSTOLAKIS: Guidelines? Whose
13 guidelines are these?

14 MR. SHEIKH: The MEB guidelines, the way
15 the plans are designed to always take the full strip
16 out.

17 MEMBER APOSTOLAKIS: I don't know what
18 that means. The guidelines take precedence.

19 MEMBER DENNING: The question is do you
20 really believe that -- I mean I realize you may not
21 believe in a guillotine break, but to say that it
22 happens in just the most ideal fashion so that there
23 aren't lateral forces, that certainly doesn't seem
24 like a very good regulatory, conservative regulatory
25 position.

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1 I can certainly see in certain conditions
2 where that's the most conservative thing to do, but
3 here it just happens that everything lined up so that
4 it's going to go and not hit anything whereas
5 something that might be more real, there would be some
6 probability it would.

7 MEMBER SIEBER: Maybe I can give you a
8 little insight. I've not seen formal experiments in
9 pipe breaks, but I've seen coal-fired power plant
10 boiler tubes burst and what they do and they actually
11 do not whip around. They will deflect into some
12 position where there is some minimization of the
13 forces on it and just stay there. They may be twisted
14 and they change from the original flow vector, 90
15 degrees or what have you, but they don't flip around
16 and spray like a firehose does. I don't know if that
17 provides any insight or not.

18 MR. SHEIKH: But going back, this is --
19 this is like defense-in-depth. Once we have
20 established that the pipe moves sideways for the
21 purpose of Generic Issue 80, even if it hits those
22 bundles, we are saying that it's not going to
23 completely block the pipe and that's the issue. We
24 can defer on whether the pipe will not whip on the
25 side, but as far as the purpose of the Generic Issue

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1 80, even if the pipe hits it, it's not going to crimp
2 it completely shut and that's the issue.

3 MEMBER DENNING: It's really not taking
4 credit for the possibility that it will whip in the
5 other direction which from that configuration wouldn't
6 matter anyway. It will hit some of the pipes
7 regardless, but if it moves, it hits.

8 MR. SHEIKH: On the GE-5 plants, there are
9 a total of four plants and these are the later
10 versions of the plants. Most of these plants have
11 installed pipe with restraints and we looked at the
12 piping analysis reports for these plants and we found
13 that * (10:52:25) point 2 is the only one predicting
14 a break as on the intermediate valve as shown on page
15 16.

16 This break.

17 Although the piping system, we looked at
18 the piping system for all the four plants that they
19 all supplied by GE, configuration is the same,
20 everything, and all other three plants don't postulate
21 a break there.

22 So if bad breaks happen, there is a pipe
23 restraint here and we are saying even if there is a
24 possibility, we don't know, but there is a possibility
25 that this pipe restraint can stop the pipe from

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1 impacting the CRD bundles here because the pipe
2 vertical section of the pipe will move here.

3 However, once it hits, it's going to hit,
4 part of it is going to hit the concrete and part of it
5 is going to hit the CRD bundles and then going back to
6 my original argument, if it happens, it won't crimp
7 it.

8 MEMBER DENNING: Before you go on, let's
9 talk about the crimping argument, just to see whether
10 -- there are some figures here that look like they're
11 real pipes, but as far as ANSYS's ability to predict
12 this, how good is it really able to do this? Isn't
13 this a -- I mean, this is a pretty difficult problem
14 as you get into the kinking area and I think that you
15 are making an argument that it failed before crimping
16 shut and do we really believe your failure criterion,
17 or is it possible that in the ANSYS analysis it's
18 believed to be a conservative assumption to say well,
19 it will fail at a certain condition whereas it could
20 be that in reality that it is able to survive to the
21 crimped position? How much confidence should I have
22 in that ANSYS' ability to predict this kind of
23 condition which gets into a kind of an unstable mode
24 when you get to a certain location?

25 MR. SHEIKH: I don't have the data here,

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1 but for another project we recently ran with another
2 program, Abacus and the results are the same.

3 MEMBER DENNING: Well, that doesn't
4 necessarily give me a lot of confidence.

5 Bill, what's your feeling? How much
6 should I believe that analysis?

7 VICE CHAIRMAN SHACK: You know, that's the
8 thing, these things always predict the deformation.
9 The failure is the tricky part. It would be nice to
10 see the analysis benchmarked against some of these
11 experiments.

12 MR. SHEIKH: Right, but you know we are
13 talking about is the threshold of failure. It's very
14 difficult to predict, but looking on the other hand,
15 we are talking about a force of a thousand kips
16 hitting these small pipes and we are saying that 6
17 percent of the force it can destroy 70 of these pipes.
18 So we have to look in the order of the magnitude of
19 the problem.

20 MEMBER DENNING: And how many -- I was
21 kind of wondering in the analysis, when we assume that
22 some are crimped, how many do we assume are crimped?
23 Does it just take one to get you into problem or do
24 you have to have multiple of the rods not able to
25 enter the core.

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1 MR. SHEIKH: Say that once again?

2 MEMBER DENNING: Does it only take one rod
3 not entering the core to get you into trouble and do
4 we -- is that what you assume?

5 MR. VANDERMOLEN: It takes more than one
6 rod, but we don't take credit for that. We assume
7 that -- I'll get into that in a minute.

8 VICE CHAIRMAN SHACK: But coming back,
9 your argument is again with a very small force you're
10 going to fail the pipe, therefore with the realistic
11 bigger force, your chances of failing the pipe are
12 virtually one and you're going to take it at point 1
13 anyway and so you're conservative.

14 MEMBER DENNING: But I do see where you've
15 got these bundled where you're running into and you
16 made
17 -- knocked the heck out of the first couple of them
18 and then as they're kind of losing energy and you have
19 to get one, if that's what it took and that's kind of
20 what I'm wondering, is how many --

21 MR. SHEIKH: That's what I'm trying to
22 say, that the force, the impact force is dependent on
23 the gap between the RCS pipe and the bundles. And
24 what I calculated assumes that it's only a 6-inch gap
25 and you hit the big pipe on the 70 bundles as 70 pipes

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1 in a bundle and it only takes 6 percent of the force
2 to destroy all those bundles.

3 MEMBER DENNING: Oh, I see what you're
4 saying. They have -- the factor there isn't very
5 important at all. That's not what's involved with
6 bringing the pipe to rest.

7 MR. SHEIKH: So you have 25-inch gap, the
8 impact force is significantly more.

9 MR. VANDERMOLEN: Well, going on to number
10 17, let me describe a little bit about how we tried to
11 turn this in a probabilistic analysis. As Abdul said
12 a moment ago, we use the four factors that came from
13 an earlier study.

14 The end state, of course, you must
15 multiply these four probabilities together. It's
16 actually a little bit more complicated than that.
17 Again, as we said earlier for our initiating event
18 frequency, we use sort of a classic value that was
19 used in NUREG 1150. We are aware that there are
20 numerous studies that are coming up with smaller
21 numbers, but we didn't at this point want to take
22 credit for them. But we are aware that they're there.

23 The next two factors are basically
24 geometrical. We put those in the analysis only unlike
25 the initiating event frequency which is typically a

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1 long normal distribution going up or down a factor of
2 10, the next two we put together a normal
3 distribution, not long normal, but normal, thinking it
4 was more appropriate for that particular parameter and
5 the way it was based. And we actually have more data
6 there than we do in the LOCA, so we can be fairly
7 certain that we can capture it that way.

8 The interesting one is the very last one,
9 RUPTPROB which is not a rupture probability, but
10 instead the probability of the pipe whip or jet
11 impingement causing CRD system failure. Now, as we
12 were discussing before in these calculations done on
13 ANSYS really say that it's not going to happen at all
14 and when you put on a distribution there, it's really
15 sort of a degree of belief how confident are you of
16 those answers and have those colleagues told us that
17 we've allowed about a 10 percent likelihood that it
18 could happen, that the calculations might not model
19 everything correctly.

20 And the way we handle that in this
21 analysis which is has been labeled a bit primitive,
22 but we think is defensible is to take an exponential
23 distribution where it comes down, it has its maximum
24 at zero, but we adjusted the exponential parameter to
25 make the mean of that distribution equal to .1. And

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1 we put those in our various codes.

2 MEMBER DENNING: Before you get to the
3 bottom line, back on the initiating event frequency,
4 how does that get apportioned among the piping. I
5 can't remember exactly how it's really done. I know -
6 - I think in WASH 1400, it was per length of pipe and
7 that wasn't a very good way to do

8 MR. VANDERMOLEN: WASH 1400 and 1150 it
9 was considered to be 10^{-4} for all the piping put
10 together. But somehow we came up with that and I
11 can't remember whether that was -- and I think there
12 were length of pipe arguments on it as opposed to
13 number of junctions or things like that.

14 MEMBER DENNING: I've seen it done both
15 ways.

16 MR. VANDERMOLEN: Yes.

17 MEMBER DENNING: Then how did you
18 apportion that?

19 MR. VANDERMOLEN: The apportioning factors
20 are actually PIPE TYPE and TYPEFRAC as Abdul just
21 discussed.

22 MEMBER DENNING: That's a plant-wide
23 frequency and then these others --

24 MR. VANDERMOLEN: Fortunately, you have
25 two of the systems that could be involved and then to

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1 the vulnerability areas near the SER new lines.

2 I should turn it over to Abdul for that.

3 MR. SHEIKH: As I explained, the other two
4 factors of PIPETYPE is the issue of the total number
5 of breaks in all the high energy lines in the
6 containment. I'm sorry, the total of the fraction of
7 the total number of breaks in the RHR lines inside the
8 containment divided by the total number of breaks and
9 the high energy lines with the steam line, the feed
10 water line, all the lines which are inside the
11 containment.

12 MEMBER DENNING: How do you decide what
13 that fraction ought to be? Is it length of piping or
14 is it --

15 MR. SHEIKH: No, the number of breaks.

16 VICE CHAIRMAN SHACK: Which is like high
17 stress junctions.

18 MEMBER DENNING: The number of junctions.

19 VICE CHAIRMAN SHACK: Well, high stress
20 locations.

21 MEMBER DENNING: High stress locations.
22 So it's proportional to the number of high stress
23 locations.

24 MR. SHEIKH: Right.

25 VICE CHAIRMAN SHACK: Which probably isn't

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1 too different from other ways you could apportion it
2 out.

3 MEMBER APOSTOLAKIS: What is a *
4 (11:02:57).

5 MR. VANDERMOLEN: WASH 1400 goes back a
6 long ways.

7 MEMBER APOSTOLAKIS: I thought there was
8 more recent work.

9 MR. VANDERMOLEN: There is more recent
10 work. They all predict lower values. We thought if
11 we used this, no one would argue.

12 (Laughter.)

13 MEMBER APOSTOLAKIS: Unless your results
14 were undesirable.

15 MR. VANDERMOLEN: Well, that's true.
16 Well, absolutely, yes. This happens fairly
17 frequently. Quite often at Engineer * (11:03:28)
18 Space, we are working in areas that are pushing the
19 envelope a little bit on PRA technology. So it's not
20 an unusual situation. We try to bound it where we
21 can. And unlike a classic PRA, we are sometimes using
22 a considerable approach, at least I would not want to
23 -- I'm not sure I would feel comfortable closing this
24 issue out by taking credit for one of these newer
25 distributions until it's been thoroughly approved,

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1 peer-reviewed and certainly with the stamp of this
2 Committee on it.

3 MEMBER APOSTOLAKIS: So your definition of
4 classic is going back to the reactor safety standard?

5 MR. VANDERMOLEN: Yes.

6 MEMBER APOSTOLAKIS: Not to the Romans and
7 the Greeks.

8 MR. VANDERMOLEN: Not to the Romans, no.
9 It went back to WASH 1400 and the reg. took it from
10 there.

11 MEMBER APOSTOLAKIS: So are we going to
12 see these distributions now?

13 MR. VANDERMOLEN: I didn't pick them out.
14 I dread to tell you how it came out, yes.

15 MEMBER POWERS: It seems to be in your
16 argument that a high energy line break is equally
17 probable at all, high stress location?

18 MEMBER APOSTOLAKIS: Yes.

19 MR. VANDERMOLEN: That's correct.

20 MEMBER POWERS: It seems to me I would
21 have hinted to take it -- saying it's either where it
22 doesn't damage to my CRD piping or not and since I
23 have no idea I would take it 50-50.

24 MR. SHEIKH: That is countered by the fact
25 that TYPEFRAC which is the ratio of the RHR piping

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1 which can affect the CRD bundles as compared to the
2 total length of RHR piping inside container.

3 MEMBER POWERS: Well, I will tell you what
4 my inherent fear is is that by breaking it down into
5 these PIPE TYPE and TYPEFRAC that you're trying --
6 you're segmenting down areas that you don't know
7 anything about and you get the classic problem have I
8 segmented it far enough, it doesn't matter what
9 probability I put in there or I will come up with a
10 new consequence or result.

11 MR. VANDERMOLEN: I like to think that we
12 don't do things like that.

13 MEMBER POWERS: So would I, but I mean --

14 MR. VANDERMOLEN: It's the best approach
15 we have. If there were -- I'm not aware of any other
16 bases. If anyone knows of one, we would be more than
17 happy to use it. I'm not aware of any.

18 MEMBER POWERS: I guess the responsibility
19 is to show sensitive you are the particulars of these
20 distributions.

21 MR. VANDERMOLEN: Yes. It's actually, for
22 the width of these distributions, it's not very
23 sensitive. Things are pretty well dominated by that
24 initial event frequency uncertainty which is a factor
25 of 10. These are not going to be anywhere near that.

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1 And of course, that last one with the exponential has
2 a long tail on it.

3 Going on to the next slide, I'll show you
4 what happens when you start cranking it through. It's
5 the usual sort of thing and I have to give the sermon
6 I always have to give whenever I show one of these
7 tables. The fact that we've shown them to two
8 significant figures does not mean that we know these
9 things to that accuracy. The accuracy is shown by
10 looking at the various columns.

11 VICE CHAIRMAN SHACK: Don't apologize,
12 this is great.

13 (Laughter.)

14 MR. VANDERMOLEN: The major reason -- I
15 always have to apologize to someone --

16 VICE CHAIRMAN SHACK: No, you don't have
17 to. This is great.

18 MR. VANDERMOLEN: One time I tried doing
19 it at just one significant figure, then the point
20 estimates looked just like the means and somebody
21 asked if I'd actually done the work.

22 (Laughter.)

23 MR. SHEIKH: Just a moment if I could
24 interrupt, we always have to make that speech.

25 (Laughter.)

1 VICE CHAIRMAN SHACK: These things are
2 critical. These guys did the right thing here.

3 MR. VANDERMOLEN: I'd like to discuss a
4 little bit what this really means and I think I can --

5 MEMBER APOSTOLAKIS: Why are your means so
6 close to the * (11:08:01)

7 MR. VANDERMOLEN: As far as I can tell
8 it's fortuitous. I think it's because the exponential
9 and the -- it's being dominated by the --

10 MEMBER APOSTOLAKIS: I think it's because
11 you have long tails. It shouldn't happen. So what
12 happened to your tails?

13 MR. VANDERMOLEN: They're there. It's
14 10,000 -- it's wiggling every one of those parameters
15 up through its distribution and we did the calculation
16 10,000 times.

17 MEMBER APOSTOLAKIS: So the mean is a
18 rigorous Monte Carlo result?

19 MR. VANDERMOLEN: Yes. Not LHS, it's
20 Monte Carlo.

21 MEMBER APOSTOLAKIS: Yes.

22 MEMBER POWERS: It simply refutes the oft-
23 quoted argument that point estimates are close to
24 medians.

25 MEMBER APOSTOLAKIS: Yes, but I think it

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1 depends a lot on the shape of the distribution.

2 MR. VANDERMOLEN: It does.

3 MEMBER APOSTOLAKIS: And also it depends
4 on what you call point estimate. What is your point
5 estimate?

6 MR. VANDERMOLEN: It's multiplying four
7 means together.

8 MEMBER APOSTOLAKIS: The means. You see,
9 the biggest question with the PRAs is whether the
10 inputs are actually means.

11 MR. VANDERMOLEN: Yes.

12 MEMBER APOSTOLAKIS: It's not so much what
13 happens to the inputs after you calculate. As someone
14 from the staff told us once, they are means because we
15 say they are which I thought was a very good answer.

16 MEMBER POWERS: When you do an uncertainty
17 analysis you take that number and you put an error
18 factor on it and it will still be the mean.

19 MR. VANDERMOLEN: It's a wider question
20 than this generic issue, but --

21 MEMBER APOSTOLAKIS: The driver here,
22 that's important. Do you have a single event or
23 failure that drives these numbers? It doesn't appear
24 that you have that.

25 MR. VANDERMOLEN: No, not that I know of,

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1 no.

2 MEMBER APOSTOLAKIS: Yes, you don't.
3 Okay, so you really have defense-in-depth too.

4 MR. VANDERMOLEN: I guess we do.

5 MEMBER APOSTOLAKIS: You don't guess,
6 however, you do.

7 (Laughter.)

8 MR. VANDERMOLEN: I hadn't thought about
9 it.

10 MEMBER APOSTOLAKIS: Well, that's why you
11 come before this Committee to get insights.

12 (Laughter.)

13 MR. VANDERMOLEN: Well, I have to say
14 appearing here is often very thought-provoking.

15 MEMBER DENNING: And this is the sequence
16 frequencies. This is not the core damage frequency.

17 MR. VANDERMOLEN: That is exactly what I'm
18 about to address. It's very tempting to call these
19 core damage frequencies, but it isn't really. Calling
20 these core damage frequencies -- in a very real sense
21 we do, is a very conservative assumption and I think
22 I'm going to try and address -- I can't remember the
23 exact reading of your question earlier, but let me see
24 if I can address it.

25 What happens in this reactor when this

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1 actually happens, you've broken a pipe. The plant,
2 the reactor depressurizes over a period of time, not
3 instantly, but fairly rapidly. The chain reaction
4 stops very quickly due to the high voiding, but once
5 you've gotten the whole core steam blanket, it starts
6 refilling, some of that I sort of left out.

7 Now it is true that a voiding water
8 reactor at cold, clean, beginning-of-cycle conditions
9 can achieve criticality on just two rods, if they're
10 adjacent, or diagonally adjacent. Now this is clearly
11 going to be a troublesome situation, but ultimately
12 you want to take this reactor apart after the
13 accident. You've got to have to find some way of
14 getting it subcritical. It's not impossible, but it
15 is going to be troublesome.

16 But look at what happens right afterwards.
17 As it comes in, you are going to get plenty of
18 voiding. You will refill the core up to the collapse
19 level, up to the two-thirds core height of this
20 accident that matches the jet pump height. And it
21 takes 30 to 40 seconds to refill that core. I will
22 say that because I don't want to beat this to death,
23 but the original question of this Generic Issue was
24 the possibility of reactivity excursion and the
25 reactivity excursion -- the fill time constant of the

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1 fuel is about six seconds, less for some of the modern
2 designs, but on that order.

3 You are not going to add reactivity so
4 quickly that you don't get moderator feedback. So
5 you're going to lower the rods by the voiding and
6 you're not going to get that excursion.

7 You do worry a little bit about
8 overheating certain areas of the core and it's part of
9 the reflood, but maybe not as much as you might think
10 because you will turn the chain reaction back on again
11 when you're quenching the core, so you're already
12 turning it over.

13 But you can get into trouble. This is not
14 a benign event. Ultimately, you're going to reflood
15 that core and it's not going to shut off. Now
16 reflooding the core keeps you from disaster right at
17 the beginning, but ultimately you have to have a heat
18 sink established to the outside. You did that in a
19 boiling water reactor with the RHR heat exchangers.
20 There are four of them and typically they will add up
21 to about two and a half percent of radiothermal power
22 in their heat dissipation capacity.

23 MEMBER APOSTOLAKIS: Decay heat.

24 MR. VANDERMOLEN: Right. Now decay heat
25 is going to be fairly high right after the event, but

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1 you have plenty of heat capacity in the pool. But
2 over the long term, after about -- let's see about
3 four hours it got up to 1 percent in your decay heat
4 production, 1 percent of radiothermal power. I'm just
5 using that as a benchmark. And you can dissipate two
6 and a half percent. That's more than enough to bring
7 the plant down.

8 But if you don't turn the core off, a
9 couple of rods out are probably not going to be a
10 problem. You probably have a fairly, honestly large
11 cluster, but as long as you can stay below a few
12 percent power, you can probably handle it, but if you
13 knock out a quarter of the core, then ultimately
14 you're going to boil the suppression pool. You're
15 going to lose MPSH and your RHR injection and you're
16 going to have a problem in the core.

17 MEMBER DENNING: Are you assuming there's
18 no ink or is there some reason why it wouldn't inject?
19 Ink?

20 MR. VANDERMOLEN: Oh, standby liquid.

21 MEMBER DENNING: Standby liquid, yes.

22 MR. VANDERMOLEN: We didn't give credit
23 for it and the reason is semi-liquid control is sized,
24 well, it was originally sized to borate reactor with
25 the reactor vessel's normal inventory. Now in this

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1 accident, you are deluding it with the entire
2 suppression pool, pumping it through. Now that adds
3 about 7.5 vessel inventories, at least at Browns Ferry
4 which is the plant I had them check. I think the
5 others are going typically to that. So that's going
6 to lower your ultimate concentration by roughly a
7 factor of eight. It's not going to be enough to bring
8 you some critical -- now that, I don't have any
9 numbers. I understand that some of these plants are
10 now using isotopically enriched boron 10. I don't
11 know what these situations will be there. But that's
12 the reason we're not giving credit for it.

13 Nor have we given credit for another
14 possibility, every boiler has some way of pumping
15 river water in there. Usually, it's a chain of valve
16 or two between the service water and the RHR
17 injection, but there's always some way where you can
18 ultimately flood the whole thing. It's not normally
19 credited for something like this, but that could be
20 done as well. That would be manual operation of the
21 part of the operators unlocking padlocks and what not
22 on valves or putting in flanges, * (11:15:17) pieces
23 between flanges, something of that nature. You
24 clearly would never want to do that under any normal
25 circumstances, but they do that have. It's called --

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1 what is it called? I'm having a senior moment here.

2 MEMBER DENNING: The ultimate disaster.

3 MR. VANDERMOLEN: Well, they have some
4 slang words for it that I don't want to repeat here,
5 but I can't remember the polite word for it.

6 Standby coolant supply, I believe it is,
7 is what you find in the training manuals. So that is
8 also there.

9 Well, keeping that all in mind, you see
10 that what we've calculated here are estimated, a
11 probability or a frequency of that state where you
12 have a refilled reactor with some number of rods left
13 behind. That does not necessarily equate to a core
14 damage frequency. However, our thresholds for core
15 damage frequency are even for a plant that's fairly
16 high in its existing core damage frequency are in the
17 order of -- the threshold is 10^6 . All the plants
18 affected here in their IPEs are reporting an existing
19 core damage frequency lower than that. These are all
20 numbers that in Management Directive 6.4.

21 Normally, they'd have to have something --
22 or 10^{-5} in order to be able to take action based on
23 core damage frequency. So this isn't going to make it
24 over the threshold for that. That doesn't mean that
25 we like the situation. It means that we don't have

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1 enough basis to actually take the regulatory action
2 where the burden of proof is on the agency. We did --
3 we were not satisfied with just this. We thought
4 well, this is not really an early core damage event,
5 but let's at least look at public risk. And we had no
6 easy way of doing that because we have no plant damage
7 state for the Level 2 and Level 3 analyses, but what
8 we did, there was one that had some similarities and
9 that is a plant damage state called PDS7 in the NUREG
10 50 analysis of the Peach Bottom Plant. This one was
11 initiated by an inadvertently opened relief valve that
12 meets at the suppression pool.

13 I'm not going to go into the -- all the
14 details of that. You have an expert sitting right
15 over there who knows all about ATWS events, but you
16 wind up in a situation where again, you have a reactor
17 that isn't shutting off and a heated up pool. It is
18 different in that this PDS involves the possibility of
19 high pressure in the vessel, whereas in this generic
20 issue since it's started by a large break LOCA, you
21 know that the vessel will be depressurized.

22 We have a code that basically uses tabular
23 information to reproduce the NUREG 1150 Level 2
24 analysis and we ran that out and using the Generic
25 Issue standard site, which is not the same as Peach

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1 Bottom, it's a uniform population density 340 people
2 per square mile which is the U.S. average that we use
3 for these Generic Issues. And we got about .89 person
4 per rem per reactor year out of this. Obviously, I
5 don't believe the .89. This is a different plant
6 damage statement about that being accurate anyway, but
7 somewhere in the order of 1 person rem per reactor
8 year or less which is well below our threshold.

9 And I didn't put it on the slide, but just
10 for the fun of it, we also ran the calculation since
11 we had the computer set up with a classic LOCA plant
12 damage state, that's PDS-1. It got a risk value in
13 terms of person REM per reactor year that was actually
14 less in the order of .6.

15 So based on that --

16 MEMBER DENNING: When you said classic
17 LOCA, did you mean leading to core damage, or did you
18 just mean LOCA? A mitigated LOCA?

19 MR. VANDERMOLEN: Well, for that plant
20 damage state in the NUREG 1150 that is a LOCA where
21 the ECCS didn't work, so that damage state does assume
22 that you are melting the core, yes.

23 So ultimately, based on all this, that
24 although we do intend to keep an eye on these
25 configurations in the future, I don't like surprises

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1 like this, but based on these numbers, the core damage
2 frequency and public risk are below our thresholds for
3 taking action, and if you gentlemen agree and will
4 give us a letter to that effect, we intend to close
5 this out with no additional requirements.

6 I should note in passing that our
7 experience has been that the industry does pay
8 attention to these Generic Issues even when they are
9 closed out and I suspect that there may be more
10 attention placed on the inspection of those vulnerable
11 sections of piping, maybe a little bit extra as a
12 result of this, but I can't really take credit for
13 that.

14 That concludes our presentation, so
15 gentlemen, I and Mr. Sheikh are more than happy to
16 answer any of your remaining questions.

17 MEMBER BONACA: I have a question about
18 the four configurations you looked at, different
19 design. How comfortable are you that those are pretty
20 much also the piping configurations and that that will
21 be common to all of them. Are they pretty standard?

22 MR. SHEIKH: They're pretty standard, but
23 they are supplied by GE plants and the previous
24 walkdowns performed by -- for NUREG previously in in
25 1998 determined that to be true.

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1 MR. VANDERMOLEN: But an addendum on that,
2 I discussed that rather extensively with Admiral Hare
3 * (11:21:13) because erratically I'm the one that
4 wrote the analysis back in 1983 and felt a little bit
5 stung by this revelation, so I -- based on that, on
6 the walkdowns, I was willing to agree with him, but I
7 wanted to make pretty sure.

8 MEMBER SIEBER: All right, any additional
9 questions?

10 If not, I'd like to thank you, Abdul and
11 Harold and Jack for the presentation today and since
12 there are no further questions, Mr. Chairman, I turn
13 it back to you.

14 VICE CHAIRMAN SHACK: Ahead of schedule
15 again. We will recess for lunch until 12:45.

16 (Whereupon, at 11:22 a.m., the meeting was
17 recessed, to reconvene at 12:45 p.m.)

18 VICE CHAIRMAN SHACK: We'll come back into
19 session now.

20 Our next topic is Resolution of ACRS
21 Comments on the Draft Final Regulatory Guide entitled
22 "Risk-Informed Performance-Based Fire Protection for
23 Existing Lightwater Nuclear Powerplants." And George
24 will lead us through this topic.

25 MEMBER APOSTOLAKIS: Thank you. Our

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1 Subcommittee on Fire Protection reviewed this matter
2 during its May 17, 2005, meeting, and the full
3 committee reviewed it during its 523rd meeting last
4 June, at which meeting we wrote a letter to the EDO
5 dated June 14, 2005.

6 And in the letter we had six
7 recommendations, the most important one being the
8 first recommendation that the Regulatory Guide should
9 not be issued in its present form, and there were
10 other comments, conclusions, and recommendations.

11 We received a response from the EDO in
12 August of this year, in which the staff states that
13 they agree with our -- with five of our six
14 recommendations, and they disagree with the last one,
15 which was that the Regulatory Guide should be revised
16 to provide definitions of the maximum expected fire
17 scenario and limiting fire scenario that are
18 acceptable I guess to us.

19 So the staff disagreed with that. I think
20 the main reason was that these definitions had already
21 been given in NFPA 805, which is an approved document.
22 And we never got back to approve documents and amend
23 them, do we?

24 So that's where we are now. I understand
25 today's session will be a relatively short one. And

1 we were ready to write a letter, but I guess we will
2 not. So I will turn it to Mr. Sunil Weerakkody of the
3 Office of Nuclear Reactor Regulation, who will lead us
4 through this. Am I doing the right thing?

5 MR. LYONS: Actually --

6 MEMBER APOSTOLAKIS: I am not doing the
7 right thing. I'm turning it over to Mr. Lyons.

8 MR. LYONS: Yes. This is Jim Lyons. I'm
9 the Director of the Division of Systems Safety and
10 Analysis, and I just wanted to say you are coming back
11 with the -- with where we are on this Reg Guide. We
12 had hoped to have it all finalized and able to bring
13 back a completely revised version.

14 We still have a few things that we're
15 working on that we'll go through today, so we can't
16 give you the final. But I think we can give you a
17 good idea of where we're going and what we're doing.

18 The other thing I wanted to say is, maybe
19 you've seen the NRR is going to be reorganizing. And
20 in the new reorganization I'm going to be the Director
21 of the Division of Risk Assessment, and so we'll have
22 all the -- you know, Mike Tschilz, currently the
23 branch of SPSB, will be in my division.

24 But included in our division will also be
25 the Fire Protection Branch. So Sunil will be coming

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1 over and will be working with us in the Division of
2 Risk Assessment. So we're going to continue to move
3 forward in risk-informing and using performance-based
4 regulations in the fire area. And so I just kind of
5 wanted to let you all know that as you move forward.

6 MEMBER APOSTOLAKIS: When do you think you
7 will come back requesting a letter?

8 MR. LYONS: We are looking at -- well,
9 we'll go to the next last slide first, I guess, which
10 is really December we would have the product ready to
11 come to you. So I think it would be the first of next
12 year that we would be --

13 MEMBER APOSTOLAKIS: February.

14 MR. LYONS: -- February that we would be
15 coming back to finalize this.

16 MEMBER APOSTOLAKIS: I understand we have
17 two persons on the phone. Would you please identify
18 yourselves?

19 MR. EUTRISS: Tom Eutriss from EPM.

20 MEMBER APOSTOLAKIS: Just one person,
21 then?

22 MEMBER KRESS: Must be.

23 MEMBER APOSTOLAKIS: And what does EPM do
24 related to the subject matter of this meeting?

25 MR. EUTRISS: We are a fire protection

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1 engineering consultant.

2 MEMBER APOSTOLAKIS: Thank you.

3 Okay. Now we'll go to Sunil.

4 MR. WEERAKKODY: Yes.

5 MEMBER APOSTOLAKIS: All right.

6 MR. WEERAKKODY: Again, my name is Sunil
7 Weerakkody. I am the Section Chief of Fire
8 Protection.

9 Next slide, please.

10 As Jim mentioned, we came to you about two
11 months ago to ask your endorsement on the Regulatory
12 Guide for 805 in its final form. You had a number of
13 comments. One major comment was to not issue the Reg
14 Guide in the form in which we presented it to you.

15 Since then, we have spent about two months
16 discussing your comments. We had a public meeting to
17 share your comments with the other stakeholders, all
18 external stakeholders.

19 MEMBER APOSTOLAKIS: Geez. Do we have
20 that much of an impact?

21 MR. WEERAKKODY: In this particular case,
22 you did.

23 MEMBER APOSTOLAKIS: Oh.

24 MR. WEERAKKODY: Subsequently, we made
25 some changes to the Regulatory Guide. NEI made

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1 several significant changes to NEI 04-02 to address
2 your comments.

3 Activities on 805 will conclude on other
4 risk-informed regulations such as proposed rule on
5 10 CFR 50.46(a). As a result, we identified several
6 other issues that we must address, which is why we are
7 -- you are not seeing the final product today.

8 At the conclusion of today's presentation,
9 if time permits, at -- I plan to provide you at a very
10 high level what those issues are. Today, we are not
11 going to seek your endorsement to issue this Reg
12 Guide. We want to inform you of the changes that we
13 made to the Reg Guide and the NEI-04 to address your
14 six comments.

15 After we address all issues I mentioned
16 about, we will submit the Reg Guide and NEI report,
17 too, for your review and endorsement around -- in
18 December.

19 Next slide, please.

20 Today's presentation will consist of three
21 main items. First, Paul Lain, the Project Manager for
22 805, will spend a few minutes to inform you about
23 where we are with respect to the implementation of
24 805. His presentation is relevant, because he -- it
25 will go a long way in addressing a major concern that

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1 you expressed at the last meeting with respect to
2 licensee's plans to develop and use fire PRAs in
3 support of 805.

4 Then, Bob Randlinski will present to you
5 the list of your comments and how we changed NEI-042
6 and Reg Guide to address your comments to the best of
7 our understanding of those comments.

8 Next slide, please.

9 The next step, we plan to have several
10 meetings internally, and then also with the public, to
11 discuss the -- a couple of the other issues we need to
12 address. Specifically, we want to meet with our pilot
13 also and get their views. Therefore, our planned next
14 step is to provide a final Reg Guide, and NEI will
15 forward it to you in mid-December, and seek your
16 endorsement to release it next year at that time.

17 And with that, I would like to turn it
18 over to Paul Lain.

19 Also, I just want to say we have Dr. Ray
20 Gallucci and Dr. Gareth Perry, in case you have any
21 questions that are difficult for us to answer. Okay.

22 MR. LAIN: Good afternoon. I'd like to
23 just give the committee a short brief on 805, keep you
24 guys abreast on the implementation.

25 We currently have commitments from Duke

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1 Power and Progress Energy to transition all of their
2 12 units to 805. We have been informed that Dominion
3 and Constellation are considering transitioning their
4 fleets. And single plants like Beaver Valley and
5 Calloway are seriously considering transitioning.

6 We expect in December that we'll -- it'll
7 probably be a decision point for a lot of facilities,
8 since there's a deadline enforcement discretion for
9 existing non-compliances that ends in December 31st.

10 We have chosen Oconee and Dennis Hennike
11 from Duke, and Sharon Harris from Progress to be our
12 pilot plants, and we had a kickoff meeting with them
13 in August to share some schedules. And we're going to
14 meet with both of them in November for our first pilot
15 observation to review their evaluations of fire-
16 induced multiple spurious circuit failures, nuclear
17 safety performance criteria, and the change control
18 process that they're going through.

19 Our second visit right now is tentatively
20 scheduled in March to review their progress on how
21 they're transitioning over the fundamental elements
22 within -- it's the Chapter 3 of 805. And then, also
23 their fire PRA status, how that's coming.

24 Next slide, please.

25 With this slide, we'd like to really

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1 stress that the transitioning plants --

2 MEMBER APOSTOLAKIS: Excuse me.

3 MR. LAIN: Yes, sir?

4 MEMBER APOSTOLAKIS: How long will this
5 transition take?

6 MR. LAIN: Duke Power has committed to do
7 it in two years, and Progress Energy would like to
8 have three years to do it.

9 MEMBER APOSTOLAKIS: It takes three years,
10 huh?

11 MR. LAIN: And it's all in a lot of their
12 tracing cables and developing their fire PRA.

13 MEMBER APOSTOLAKIS: I see. So they --
14 well --

15 MR. LAIN: And what they're doing is
16 they're staggering their plants to do it, so they're
17 -- they're sort of starting a new one every -- every
18 year, and so they're --

19 MEMBER APOSTOLAKIS: The transition itself
20 does not require a fire PRA, right? It's afterwards
21 that --

22 MR. LAIN: It's afterwards that helps
23 them. But within their change control process --

24 MEMBER APOSTOLAKIS: Absolutely.

25 MR. LAIN: -- a fire PRA really helps

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1 them, if they come upon areas that they do not -- are
2 not in compliance --

3 MEMBER APOSTOLAKIS: That's right

4 MR. LAIN: -- it helps them with their
5 transition. So they are working --

6 MEMBER APOSTOLAKIS: So they have already
7 started this?

8 MR. LAIN: Yes.

9 MEMBER APOSTOLAKIS: Very good. Very
10 good.

11 MR. LAIN: Okay.

12 MEMBER APOSTOLAKIS: Do we know why? I
13 mean, why did they decide to do it? I mean, the local
14 people are saying that they have invested so much in
15 Appendix R compliance. What is --

16 MR. LAIN: I think that one of the big
17 motivators is the circuit analysis. Duke Power kind
18 of did a circuit analysis process by a process of
19 elimination, short of -- I don't -- to best describe
20 that is that they sort of -- they figured out what
21 wasn't in a room, and then they -- they figured that
22 their cables were safe.

23 Now they're actually running through and
24 tracing all their cables and making sure that, you
25 know, they don't end up having a Train A and Train B

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1 affected at the same time. A lot of this came out of
2 the testing that NEI did on spurious actuation of
3 cables when the industry indicated that they didn't
4 think that more than one spurious actuation could
5 happen at a single time. And the testing ended up
6 kind of proving that the multiple spurious could
7 happen.

8 MEMBER APOSTOLAKIS: I see.

9 MR. LAIN: And so there has been a lot of
10 activity in the last couple of years to sort of come
11 to light on what the agency expects. And we've had a
12 regulatory information summary -- one or two -- we've
13 revised those -- that have come out. And these two
14 plants are two of the plants that figure they really
15 need to go back and rereview their safe shutdown
16 analysis.

17 MEMBER APOSTOLAKIS: Okay. Thanks.

18 MR. LAIN: So they've both committed to
19 spend sort of thousands of hours to sort of -- to
20 transition the tracer cables and enhance their fire
21 PRA. And I think Progress Energy quoted to do their
22 -- all their sites \$40- to \$60 million. So they've
23 committed to spend quite a bit of money.

24 Our first -- our current enforcement
25 discretion period is two years. Progress has

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1 requested that we -- we look at -- they'll need three
2 years to develop a quality fire PRA, and so we are
3 currently reviewing with the Office of Enforcement to
4 -- that request.

5 We stressed to the licensees at last
6 month's NEI information forum that it would be
7 impractical to transition without a quality fire PRA,
8 and we will be scrutinizing the ones without one, you
9 know, through the inspection process.

10 Our last item we'd like to relay I guess
11 is that we've been revisiting the PRA and the fire
12 modeling guidance, such as Reg Guide 1.174, the draft
13 guide 1.200, RES's fire PRA method -- methodology.
14 And to use for the NRC review -- and we've identified
15 that sort of a fire PRA peer review methodology is
16 needed. So I think we've been in discussions with NEI
17 on development.

18 MEMBER APOSTOLAKIS: That's interesting.

19 MR. LAIN: And so as part of our having a
20 quality fire PRA, we're working towards having a peer
21 review methodology.

22 MEMBER APOSTOLAKIS: Do you think five is
23 going to play a role in all of this?

24 MR. LAIN: I think five is one of the
25 methods. They have revised five in the -- in the Reg

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1 Guide, and we are looking at that through a fire model
2 effort. But we expect them to not just rely on the
3 old IPEEE items. We expect them to sort of advance
4 and get -- produce better --

5 MEMBER APOSTOLAKIS: Well, it originally
6 was intended to be a screening approach. So now it's
7 not screening anymore.

8 MR. WEERAKKODY: To the best of my
9 knowledge -- and Ray might be able to -- I don't
10 believe people could have 805 and have five that they
11 prepared for IPEEEs. And to the best of my knowledge,
12 no one is even planning to do that.

13 DR. GALLUCCI: This is Ray Gallucci.
14 There is debate right now on the fire PRA Standard
15 Writing Committee as to whether a five even qualifies
16 as a category 1. It's an ASME standard. It would be
17 IPEEE quality. Yes, it's the same type of thing for
18 the ASME standard.

19 MR. LAIN: So now I would like to turn it
20 over to Bob to discuss more about the specific Reg
21 Guide.

22 MR. RANDLINSKI: Good afternoon. My name
23 is Bob Randlinski. As Sunil mentioned, my
24 presentation is going to review the comments that we
25 received from the ACRS on the 805 Reg Guide, and on

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1 NEI 04-02, and talk about our response to those
2 comments, and also describe some of the changes that
3 we made to these two documents as a result of those
4 comments.

5 The first comment, as George mentioned,
6 was that they didn't believe that -- the committee
7 does not believe that the Regulatory Guide is ready to
8 be issued in its present form. We've -- we are
9 accepting the comments that were made, the specific
10 comments that were made, on the Regulatory Guide, and
11 both NEI and the staff have incorporated those
12 comments in a revision to the Reg Guide and to 04-02.

13 So hopefully, based on our presentation
14 today and our discussion, that the committee will
15 agree that the Reg Guide is ready to be issued.

16 We then plan to issue the Reg Guide next
17 year, as Sunil mentioned, after submitting a draft
18 final version to the committee in December.

19 First specific comment was that the
20 initial fire modeling approach should not be used as
21 an alternative to estimates of changes in CDF and
22 LERF. The way we addressed this was to revise
23 Figure 5-1 in NEI 04-02 and --

24 MEMBER APOSTOLAKIS: Oh, it's here.

25 MR. RANDLINSKI: It's in your handout. We

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1 can --

2 MEMBER APOSTOLAKIS: Yes. Oh, that's the
3 old one. Yes. They are burning in that -- ah. You
4 guys are so good. Oh, we have a laser point?

5 MR. RANDLINSKI: Okay. This was the area
6 of concern last time.

7 MEMBER APOSTOLAKIS: Wait, wait, wait.
8 The Reporter has a problem.

9 MR. RANDLINSKI: Okay. The area of
10 concern is -- this is the plant change evaluation of
11 the process in schematic form. And this is from NEI
12 04-02. It's Figure 5-1. This was the previous
13 revision, Revision 0, which is covered up by that
14 five. The area of concern was this path here, which
15 is the approach -- the fire modeling approach to
16 evaluating a change, and it was shown as a parallel
17 path along -- in parallel with the risk assessment
18 path.

19 And the concern was, by the committee,
20 that a change could be evaluated using this path only,
21 and you would complete the evaluation without actually
22 evaluating the risk, evaluating CDF and delta LERF.

23 MEMBER APOSTOLAKIS: That's right.

24 MR. RANDLINSKI: Okay? There was -- there
25 were words in the document itself that prohibited you

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1 from doing that, but it wasn't clear in the diagram,
2 so we changed the diagram.

3 So that same area of the schematic is here
4 under risk evaluation, and you see there's no parallel
5 path. Everything comes straight through. Everything
6 goes down through this step. Everything goes down
7 through the step where you have to evaluate delta CDF
8 and delta LERF for every change.

9 MEMBER APOSTOLAKIS: Good.

10 MR. RANDLINSKI: Okay?

11 MEMBER APOSTOLAKIS: Very good.

12 MR. RANDLINSKI: We've taken all -- we've
13 cleared all statements from 04-02 that indicated that
14 you might not be able -- or that you might be able to
15 use a fire modeling approach by itself, and included
16 some statements that made it clear that you do have to
17 evaluate risk as well as looking at the fire model, if
18 you use that approach.

19 MEMBER APOSTOLAKIS: Okay.

20 MR. RANDLINSKI: Are there any questions
21 about the figure? No? Okay.

22 The next comment was that the staff should
23 not endorse methods for evaluating delta CDF and delta
24 LERF that are not based on fire PRA. 10 CFR 50.48(c),
25 the rule, revised rule, and the NFPA 805 do allow risk

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1 assessments to be performed without a full fire PRA.

2 So we cannot require the licensees to
3 develop a full fire PRA and use a full fire PRA.
4 However, to the extent possible, we encourage the
5 licensees to do this and --

6 MEMBER APOSTOLAKIS: Well, I would take a
7 different approach, Bob. I would say, you know, you
8 show me a delta CDF and a delta LERF, I want to be
9 convinced that this is a real delta CDF and a real
10 delta LERF. I don't know how -- I don't care how you
11 do it. Why should I care whether they have a full
12 fire PRA or a 63 percent fire PRA? Maybe, you know,
13 you don't need a full fire PRA in some instances.

14 MR. RANDLINSKI: Great. Then we're in
15 agreement.

16 MEMBER APOSTOLAKIS: But the focus is
17 delta CDF and delta LERF. In other words, I don't
18 think anyone should come here -- or to you, actually,
19 not to us -- and say, "We calculated delta CDF, and we
20 didn't have a full fire PRA. And, you know, I think
21 it's okay."

22 The question is: is your delta CDF
23 realistic? That really should be the focus -- how you
24 did it. I mean, some people are maybe so gifted that
25 they can just do it without any calculations. It's

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1 three 10⁻⁵. You know? But if it's real, then it's
2 real.

3 MR. RANDLINSKI: Good. Then we're in
4 agreement.

5 MEMBER APOSTOLAKIS: So I would -- we are
6 in agreement, but maybe the tone -- the reason why
7 we're in agreement may be different. I think it would
8 be nice to emphasize that when you evaluate delta CDF
9 and delta LERF, you go back to Regulatory Guide 1.174,
10 and you follow the rule. It says, you know, you
11 should do this.

12 MR. RANDLINSKI: Right.

13 MEMBER APOSTOLAKIS: Should represent --
14 certain decisions, you know, represent, you know,
15 everything you can think of and all that.

16 MR. RANDLINSKI: Right. And we do
17 reference Reg Guide 1.174 --

18 MEMBER APOSTOLAKIS: Okay.

19 MR. RANDLINSKI: -- for that purpose.

20 MEMBER APOSTOLAKIS: But to go into what
21 50.48(c) and NFPA allow, yes, I mean, they allow it.
22 But if your delta CDF is not realistic, I'm sorry.

23 MR. RANDLINSKI: Okay. And as Paul --

24 MEMBER APOSTOLAKIS: Right.

25 MR. RANDLINSKI: -- as Paul mentioned, the

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1 first two utilities that are adopting 805 are doing --
2 are developing full fire PRAs.

3 MEMBER APOSTOLAKIS: Which is --

4 MR. RANDLINSKI: And we really anticipate
5 or expect that all of the utilities will do that.

6 MEMBER APOSTOLAKIS: Of course. I mean,
7 you are switching, and I think Paul mentioned, what,
8 \$50-, \$60 million that was spent. It's ridiculous to
9 do it half --

10 MR. LAIN: Duke Power has kind of said
11 that, you know, you -- for the cost of a fire PRA, you
12 know, it's like doing -- doing a partial three times
13 over. You know, you might as well do the full fire
14 PRA to get the economy --

15 MEMBER APOSTOLAKIS: Exactly.

16 MR. RANDLINSKI: -- and get the payback --

17 MEMBER APOSTOLAKIS: Exactly.

18 MR. RANDLINSKI: -- in the future, and to
19 be able to do the change control process
20 efficiently --

21 MEMBER APOSTOLAKIS: Absolutely.

22 MR. RANDLINSKI: -- that you might as well
23 just make it --

24 MEMBER APOSTOLAKIS: I don't know. Does
25 anybody know? Ray, maybe you know. How much does the

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1 -- a full fire PRA cost?

2 DR. GALLUCCI: When I was at Ganay,
3 starting with an internal events fire PRA, and already
4 having the cables traced, it cost about \$150K.

5 MEMBER APOSTOLAKIS: That's nothing.

6 MR. LAIN: The big cost is tracing the
7 cables, which has been said 5- to 7,000 manhours.

8 MEMBER APOSTOLAKIS: So let me understand
9 this. If they don't do a full fire PRA, they don't
10 have to trace the cables?

11 MR. LAIN: I mean, I would think they
12 would need to trace the cables for -- you know, in
13 that area that they're doing the change in.

14 MEMBER APOSTOLAKIS: Right.

15 MR. LAIN: And that's a big cost.

16 MEMBER APOSTOLAKIS: That's the point. I
17 mean, it's not just what PRA wants.

18 MR. LAIN: Right.

19 MEMBER APOSTOLAKIS: What's the
20 alternative? So one way or another they would have to
21 do it. Maybe not in a complete case -- sorry?

22 MEMBER POWERS: They would have to do it
23 to make a change. But if they don't do it, there
24 could be some latent defect in there -- in the routing
25 that could cause a problem.

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1 MEMBER APOSTOLAKIS: In general --

2 MEMBER POWERS: I guess that's what the
3 inspections are designed to find out.

4 MEMBER APOSTOLAKIS: I think we are past
5 the time when, you know, we could do 30 percent of a
6 PRA and a little bit of the fire PRA. I mean, if you
7 want to have risk-informed decision-making, you'd
8 better have the tools. And I think this is very good.
9 I mean, you know, you have to have the PRA, the fire
10 PRA, because in the past, you know, since '98 when the
11 Regulatory Guide came out, I mean, I think the staff
12 has gone out of its way to accommodate incomplete
13 PRAs.

14 You know, and if you don't have a Level 2,
15 look, maybe you can do this, you can do that, dance a
16 little bit. I mean, you are okay. If you don't have
17 a shutdown PRA, maybe you can -- well, maybe it's time
18 now to say, "No, you should."

19 MR. RANDLINSKI: Okay.

20 MEMBER APOSTOLAKIS: That's why people say
21 that sometimes these committees pontificate.

22 (Laughter.)

23 MR. RANDLINSKI: Okay. The next comment
24 was very similar. The comment was that NEI 04-02
25 contains many statements that are inconsistent with

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1 the Commission's policy of promoting the use of PRA
2 methods. In the Reg Guide, the staff should make it
3 clear that it does not endorse such statements.

4 As I mentioned, 04-02 was revised,
5 particularly in Appendix J, in Section 5.3, to make it
6 -- to encourage licensees to use a detailed
7 calculation approach to assessing delta CDF and delta
8 LERF. Also, in the Reg Guide we don't specifically
9 endorse non-PRA methods, and we do talk about PRA
10 methods.

11 Next comment was the staff should ensure
12 that parts of NEI 04-02 that endorses use correct
13 methodology and language. Sunil mentioned earlier we
14 had a -- held a public meeting with -- to share the
15 ACRS comments with NEI and discuss how we should
16 approach those comments, and which of the two
17 documents should be revised to address the comments.

18 We held several follow-up phone calls with
19 NEI. We've been working pretty closely with them to
20 fine tune their document, as well as make any changes
21 that we needed to the Reg Guide.

22 And as we got revisions to 04-02, we had
23 full review of those by members of the staff, fire
24 protection, also in the research group, to review the
25 fire modeling and the PRA portions of it. And we

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1 believe that the methodology and language that's now
2 used in 04-02 is correct.

3 The next part of the presentation is to
4 give you a little more specifics on how we've changed
5 each of the documents. With respect to the Reg Guide
6 -- again, general comment, we agree with your
7 comments, and we incorporated those comments in the
8 final draft.

9 The Reg Guide states that risk evaluations
10 for non-screened changes should use PRA methods and
11 tools. We added PRA quality references, including Reg
12 Guide 1.174, Reg Guide 1.200, and the ANS fire PRA
13 standard. And we also noted that future additional
14 guidance for fire PRAs will be issued, and it will be
15 -- that future guidance will follow those reference
16 documents.

17 MEMBER APOSTOLAKIS: So, Ray, you
18 mentioned the ANS fire PRA standard. Can you tell us
19 in 30 seconds what the status of that is?

20 DR. GALLUCCI: The current status -- we
21 had a Writing and Review Committee meeting at PSA 05
22 a couple of weeks -- a couple of weeks ago in San
23 Francisco. Comments from -- I think ANS comments had
24 -- preliminary ANS comments had been received.
25 Comments were received from the various reviewers, and

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1 the Writing Committee is preparing what will be a
2 draft for concurrent public comment and ANS's Risk-
3 Informed Subcommittee review probably the end of this
4 month or sometime next month.

5 By the end of -- certainly by Thanksgiving
6 the final draft should be out for public comment and
7 ANI Risk Committee review. So it's probably within a
8 year of completion at that point.

9 MEMBER POWERS: Ray, is that standard
10 going to include fire during shutdown conditions?

11 DR. GALLUCCI: It does not -- it doesn't
12 specifically give any -- it's an at-power type of
13 standard as the other ones. So it won't have anything
14 specific for fire at shutdown.

15 MEMBER POWERS: Isn't fire -- isn't the
16 probability of a fire more likely under shutdown
17 conditions than operational conditions?

18 DR. GALLUCCI: There is different types of
19 fires that you would see under shutdown conditions.
20 I think that there's other efforts going on where
21 they're trying to -- between ASME and ANS where
22 they're trying to coordinate all of the standards that
23 are being developed.

24 And I don't know if the decision has been
25 made yet whether fire during shutdown/flood during

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1 shutdown should be part of the shutdown standard
2 itself, or whether it should be part of the fire or
3 the external event standards.

4 MEMBER POWERS: It seems to raise the
5 question of suppose someone came, said, "I'm going to
6 design my fire protection system based on NFPA 805,"
7 and he hasn't addressed fire protection during
8 shutdown?

9 DR. GALLUCCI: NFPA 805 does require that
10 fire during shutdown be considered. But the standard
11 is not going to develop any specific technical
12 requirements at this point.

13 MEMBER POWERS: So how does it work with
14 respect to this?

15 MEMBER APOSTOLAKIS: So, yes, that's a
16 good point. Bob, when we talked earlier about the
17 full fire PRA, did we include shutdown mode?

18 MR. RANDLINSKI: Do you mean in the Reg
19 Guide?

20 MEMBER APOSTOLAKIS: No. No, I mean --
21 let's go back a couple of slides. I mean, there was
22 some statement there that they have to -- no, back.
23 That they will have to use a full fire PRA. Here.
24 Assessments to be performed without a full fire PRA.
25 Does that include all the operating modes of the

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1 plant?

2 MR. WEERAKKODY: It does not include --

3 MEMBER APOSTOLAKIS: Shutdown?

4 MR. WEERAKKODY: -- shutdown.

5 MEMBER APOSTOLAKIS: But why not? I mean,
6 I think the issue is very relevant.

7 MR. WEERAKKODY: The shutdown risk is
8 relevant, George, but -- and I'm glad we have other
9 people here, but in terms of the shutdown risk, both
10 for internal events or fire we are not at the state
11 where we are capable of doing that type of evaluation.

12 Do you want to add anything, Gareth? I
13 mean --

14 MEMBER APOSTOLAKIS: So how -- so
15 presumably, then, the shutdown fire issue will be
16 handled in a different way, not probabilistic way? I
17 mean, it has to be handled, because --

18 MR. WEERAKKODY: There are several ways to
19 handle the shutdown. And if you can think of the
20 shutdown risk management, you know, when you are in a
21 shutdown, each plant, each outage, you may have, you
22 know, different configurations. And you manage the
23 shutdown risk by evaluating the different
24 configurations and make sure that each configuration
25 is safe, rather than sing quantitative PRAs.

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1 So that's -- that's one approach of
2 managing that. But, you know, if you go a step
3 further, if you're looking at any plant, any permanent
4 plant changes, obviously each plant knows if there are
5 any systems that are only important for the shutdown,
6 such as pressurized water reactor or -- which you
7 would just use for less significance during at-power.

8 So it would be considered, but we are not
9 -- what we are saying is that it would not be
10 numerically evaluated in a -- in a CDF fashion.

11 DR. GALLUCCI: Let me add that if a plant
12 does have a low power shutdown PRA model, then
13 superimposing a fire model on top of that fire PRA
14 model is somewhat analogous to what you do with the
15 internal events at power model. You would -- you
16 basically would have -- you'd have your plant
17 operating states developed, you'd have different event
18 trees, fault trees, for the shutdown operating mode --
19 shutdown modes, and you would superimpose fire
20 initiators, etcetera.

21 There would be -- of course, there's
22 probably more dependence on manual action. So if a
23 plant -- again, the starting point to model fire PRA
24 shutdown is to have a shutdown model in the first
25 place, just like the starting point to have a fire PRA

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1 is an internal events at power model.

2 MEMBER POWERS: But, Ray, what I'm
3 wrestling with a little bit here is that NFPA 805
4 requires considering all operational states. So now
5 we're writing a Reg Guide here in which we consider
6 those states where the risk of fire is the least,
7 instead of those where it's the most, it seems to me.

8 DR. GALLUCCI: I don't know if it's --

9 MEMBER POWERS: It seems to me that the
10 likelihood of fire is greater during shutdown than it
11 is during normal operations. I may be in error on
12 that. But it seems somehow we're leaving out a part
13 of the equation. Once we're done discussing this,
14 then I'll move and ask about seismically-induced
15 fires.

16 DR. GALLUCCI: The likelihood may be
17 higher for certain types of fire, but the risk isn't
18 necessarily, because you're, of course, in a shutdown
19 mode. I'm trying to recall --

20 MEMBER POWERS: Let's see, I'm a shutdown
21 mode, which means my containment most likely is open?

22 DR. GALLUCCI: Yes.

23 MEMBER POWERS: So if I do get core
24 damage, my conditional containment failure probability
25 is one?

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1 DR. GALLUCCI: But you're very unlikely to
2 get core damage in such -- in that mode, because
3 you're depowered, you're depressurized.

4 MEMBER POWERS: Gosh. That sure hasn't
5 been borne up by the shutdown risk assessments that I
6 have seen.

7 DR. GALLUCCI: Well, Ganay did a full-
8 blown PRA -- fire during shutdown, flood during
9 shutdown -- and shutdown was the minimal of all of the
10 contributors relative to fire, flood --

11 MEMBER POWERS: But those that I have seen
12 did not show that.

13 DR. GALLUCCI: Okay.

14 MR. WEERAKKODY: Can I add something? Dr.
15 Powers, with respect to shutdown, your statement that
16 fires are more likely during shutdown is true. But
17 one of the things you've also got to factor -- there
18 are two things that needs to be factored in.

19 If you go back to the -- in fact, I had --
20 there was like 600 actual fire events in a fire
21 database that we looked at when we prepared the IPEEE
22 for our plants, you know, when I was -- I recall, in
23 fact, we put a paper together in terms of the nature
24 of the fires.

25 What you will find is during outage the

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1 fires you have are, you know -- you know, you do
2 welding, because you've got -- you are working on a
3 component. You know, a piece falls, that kind of non-
4 consequential --

5 MEMBER POWERS: Maybe I'm looking at a
6 containment penetration seal with a candle. What
7 causes the fire is where -- what it does that becomes
8 important, and you're talking about reasonably rare
9 events. I mean, to argue that all of the shutdown
10 fires are inconsequential --

11 MR. WEERAKKODY: No, I wouldn't say that.
12 No, I wouldn't say that, Dr. Powers. What I would --
13 what I would say, though, is that the issue that you
14 mentioned, which was the -- we have somebody who has
15 a candle, the second relevant aspect is when you are
16 in shutdown, you are at very low decay heat level.
17 And this is not just true for fire, but true for every
18 shutdown.

19 You are at low power levels, and that's
20 why, like Chris said, your conditional --

21 MEMBER POWERS: It's just not consistent
22 with the shutdown risk assessments that I've seen. We
23 will stipulate, yes, that heat is lower. But, gee, it
24 looks to me like the numbers I've seen for Surry and
25 Grand Gulf were commensurate with normal operations,

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1 even though the time period of shutdown was very
2 small. It was a fraction of the year.

3 Even after annualizing them, they came in
4 as -- as substantial. So now if my event frequency is
5 higher, and my core damage probability, given a
6 failure, is about the same, it looks to me like my
7 risk is higher. I don't see how it can be otherwise.

8 MR. WEERAKKODY: Well, if you go to the
9 next level of detail as to what events drive those
10 shutdown risks, you know, I can only -- you know, I'm
11 not focusing on the fire. But going to the internal
12 events for pressurized water reactors, but during --
13 the fact is that you do go through some relatively at-
14 risk evolutions during mid-loop or when you have
15 things of that nature.

16 But what is not proved is it's necessarily
17 -- when you go to the shutdown risk, you can pretty
18 much look at -- you can identify and sort of recognize
19 those items that guide risk. So even though you have
20 -- you do have more fires, that does not necessarily
21 relate to higher fire risk due to shutdown.

22 But I think, you know, we'll go back and
23 take a look at this, but, you know, what I have to do
24 is, you know -- you know, say that we are asking
25 licensees to do low-power and shutdown fire PRAs.

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1 MEMBER APOSTOLAKIS: You are?

2 MR. WEERAKKODY: We are not.

3 MEMBER APOSTOLAKIS: Oh, you are not.

4 MR. WEERAKKODY: We are not. Not at this
5 point.

6 MR. LAIN: Right now, the guidance is a
7 traditional sort of fire hazards analysis for those
8 areas, and it's kind of recognized at the -- that the
9 fire PRA at shutdown is not -- not available at this
10 time.

11 MEMBER APOSTOLAKIS: But you are not
12 explicitly stating that you are excluding shutdown
13 fire.

14 MR. WEERAKKODY: No. What I am saying is
15 that in fire PRAs, the clear message we are telling
16 the licensees is that when you adopt 805, you have to
17 do a full fire PRA on the at-power more.

18 MEMBER APOSTOLAKIS: So even Progress
19 Energy and Duke, who plan to go through this major
20 conversion, are not planning to have a shutdown fire
21 PRA?

22 MR. WEERAKKODY: Not at this time. Now,
23 one thing -- you know, Ray mentioned this. Once you
24 know where your cables are, and if you have an
25 internal event shutdown model, to go the next step is

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1 relatively simple.

2 MR. RANDLINSKI: But don't forget, 805
3 requires that the licensees meet the nuclear safety
4 criteria for all modes of plant operation. The PRA
5 may not address low-power shutdown operation, but they
6 do have to meet the safety criteria.

7 MEMBER APOSTOLAKIS: I guess -- are we
8 going to have a meeting here one of these years on the
9 fire PRA during shutdown? Or we will do it in the
10 context of the ANS standard perhaps?

11 MR. WEERAKKODY: I would think it's --

12 MEMBER POWERS: They're going to tell you
13 that they didn't do it.

14 MEMBER APOSTOLAKIS: Huh?

15 MEMBER POWERS: They're going to tell you
16 they didn't address it.

17 MEMBER APOSTOLAKIS: But they developed
18 the standard.

19 MEMBER POWERS: They developed a standard
20 that didn't apply during shutdown.

21 MEMBER APOSTOLAKIS: Oh, the standard did
22 not apply here.

23 MEMBER POWERS: They're going to say
24 somebody else will do that.

25 MR. HYSLOP: My name is J.S. Hyslop.

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1 MEMBER APOSTOLAKIS: Yes, Jay.

2 MR. HYSLOP: From Research.

3 MEMBER APOSTOLAKIS: Yes.

4 MR. HYSLOP: EPRI and Research are talking
5 about doing some work in low-power shutdown and fire
6 to -- starting in '06 to look at frequency specific to
7 low-power shutdown to quantify things and develop
8 tools further for low-power shutdown analyses.

9 MEMBER APOSTOLAKIS: So when do you think
10 you will be able to come here and tell us a little bit
11 about it?

12 MR. HYSLOP: Well, we haven't even
13 developed any bullets, any schedule yet. So I don't
14 want to get into that right now, but we're -- we're
15 talking about initiating it in '06. And after we have
16 a better sense of the program and the schedules, I can
17 -- I can tell you.

18 MEMBER APOSTOLAKIS: Okay.

19 MR. RANDLINSKI: Yes. The next slide
20 pretty much repeated --

21 MEMBER APOSTOLAKIS: Okay.

22 MR. RANDLINSKI: -- it's a repeat of
23 statements I've already made. 04-02 is revised to
24 make it clear that you can't do -- can't just use a
25 fire modeling approach. You have to do a risk

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1 assessment for each and every plant change.

2 MEMBER APOSTOLAKIS: Who developed NEI 04-
3 02? Is that a legitimate question? Or it's none of
4 my business?

5 MR. RANDLINSKI: What is that?

6 MEMBER APOSTOLAKIS: Who wrote NEI 04-02?

7 MR. RANDLINSKI: NEI and --

8 MEMBER APOSTOLAKIS: Is it appropriate to
9 ask? If it's not, tell me. I know it's NEI.

10 (Laughter.)

11 MR. MARIM: Alex Marim, NEI. We hired a
12 contractor to basically develop the document that was
13 subsequently reviewed by about a handful, maybe eight
14 utility persons who are very knowledgeable in fire
15 protection, which included representatives from Duke
16 and Progress.

17 MEMBER APOSTOLAKIS: So you can't tell us
18 who that contractor is.

19 MR. MARIM: Pardon?

20 MEMBER APOSTOLAKIS: You hired a
21 contractor.

22 MR. MARIM: Yes.

23 (Laughter.)

24 Do you wish to know the name of the
25 contractor?

1 MEMBER APOSTOLAKIS: Yes, please.

2 MR. MARIM: Oh, I'm sorry. Kleinsorg and
3 Associates.

4 MEMBER APOSTOLAKIS: Great. Thank you.

5 MR. RANDLINSKI: And 04-02 does -- there
6 is a Revision 0 also -- encourage licensees to use a
7 detailed quantitative approach in assessing risk for
8 any plant changes.

9 And the last slide has to do with
10 discussion near the end of the ACRS letter. It wasn't
11 part of the recommendations, but they are actually
12 comments and questions in this regard. Had to do with
13 fire modeling approach in the LFS versus MEFS. Okay?

14 MEMBER APOSTOLAKIS: Oh, yes.

15 MR. RANDLINSKI: And you identified some
16 statements that were -- confused logic, and you were
17 concerned about the margins that were included in the
18 fire model --

19 MEMBER APOSTOLAKIS: Right, right.

20 MR. RANDLINSKI: -- to account for
21 uncertainties.

22 The document was advised to provide some
23 clarification of the safety factors that the guidance
24 recommends are used with the fire modeling approach to
25 account for uncertainties, and they also clarified

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1 that statement that you, George, may have referred to
2 as confused logic, by a simplified approach to
3 calculating --

4 MEMBER APOSTOLAKIS: Do you use those
5 words in the --

6 MR. RANDLINSKI: Yes.

7 MEMBER APOSTOLAKIS: Geez. But the
8 definitions of the maximum expected fire scenario and
9 limited fire scenario will not be changed, right?

10 MR. RANDLINSKI: They have not changed.
11 There was quite extensive discussion of both in 04-02,
12 but that was in Rev 0. And I assume you saw it.

13 MEMBER APOSTOLAKIS: Yes, I did.

14 MR. RANDLINSKI: And, of course, as we
15 mentioned before, the definition is in NFPA 805.

16 MEMBER APOSTOLAKIS: But don't you guys
17 find it confusing, though, when the limiting fire
18 scenario definition says, "One or more inputs to fire
19 scenario are up to their limit, so that performance
20 criteria is -- are not met." One or more. I mean, it
21 gives you such freedom.

22 MR. RANDLINSKI: There is guidance --

23 MEMBER APOSTOLAKIS: It's not one limiting
24 scenario, right? You can have many.

25 MR. RANDLINSKI: There is guidance in 04-

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1 02. It mentions two, in particular, that are
2 important. And it -- it does provide some specifics.

3 MR. WEERAKKODY: Dr. Apostolakis, I know
4 that you -- you remember when you had a meeting,
5 subsequently, the full committee meeting. As I
6 recall, your underlying concern was that given that
7 there is some subjective in these definitions and
8 these ratios, the fact that there was this bypass pump
9 in fire model -- and I think what we are saying is
10 that we've taken that bypass valve. Now, that kind of
11 subjective uncertainty which is there can be brought
12 into the risk analysis.

13 MEMBER APOSTOLAKIS: Is that it, Bob?

14 MR. RANDLINSKI: That's the end of my
15 presentation.

16 MEMBER APOSTOLAKIS: All right.

17 MR. RANDLINSKI: I think Alex wants to --
18 Sunil, did you want your last --

19 MR. WEERAKKODY: No, I would rather if
20 Alex goes first.

21 MR. RANDLINSKI: Okay.

22 MR. WEERAKKODY: And then takes the rest
23 of the time. You wanted to -- Alex, you wanted to
24 make some remarks, right?

25 MR. MARIM: Yes, sure.

1 (Laughter.)

2 Yes. Alex Marim, NEI. I apologize, I
3 wasn't really prepared to do so, but I can speak to
4 Dr. Powers' question about fire evaluations during
5 shutdown conditions. Those are being conducted today
6 and will continue to be conducted.

7 It remains to be seen as we start
8 developing a fire PRA, and applying a PRA to deal with
9 fire events, whether we're going to take it to a point
10 of evaluating shutdown risk from the standpoint of a
11 PRA analysis. We're not there yet. We don't really
12 see a need to do it at this particular point in time,
13 but we may evolve to that point as -- as the standards
14 are developed, etcetera.

15 That's all I have to say. Thank you for
16 the opportunity.

17 MR. WEERAKKODY: Well, in that case, I
18 will go to my last slide. What I have listed here is
19 the high level some of the other issues that we have
20 and we are addressing. As you all know, 10 CFR 50.69,
21 special treatment, that is a risk-informed rule that
22 was completed. I can't remember which year, but that
23 is already out there.

24 10 CFR 50.46(a), the proposed risk-
25 informing part of 50.46 ECCS, it's in the proposed

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1 risk stage, and 50.48(c), which is the 805, which was
2 completed last year.

3 You know, really, we did -- because of the
4 reason I mentioned before, we recognize that as the
5 agency has to maintain compatibility among these rules
6 that have been completed and that are in process while
7 accumulating their differences and purposes. As you
8 know, each one has its own purpose. 50.69 is the
9 final report, to the best of my understanding, is
10 ISIs, ISDs, and the associated risk changes.

11 50.46(a), something -- it's to do with the
12 break size for -- for pipe break and the associated
13 risk. And 50.48(c) is on fire protection.

14 So this we worked closely with the PRA
15 Branch to ensure that all the rules and guidance
16 documents benefit from each other's development. For
17 example, we have brought consistency to the
18 terminology. If you can recall, the last time when we
19 were here, one of the things that upset you was that
20 we had terms like inconsequential, non-negligible,
21 negligible. We went back in, and we -- we addressed
22 that.

23 We are not creating any -- any new words
24 in 805. We are limiting ourselves to the words that
25 are already in 1.174. And to the best of my

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1 understanding, we are using the word "minimal" now,
2 which is equal to 10⁻⁷ frequency. So we have
3 addressed that issue.

4 However, I'm not saying everything is
5 final and everything is a done deal. There are still
6 some differences we need to address.

7 Another issue we -- we identified that
8 needed addressing was the -- with respect to the
9 quality of the PRA. In line with the Commission's
10 expectations on the phased approach to quality, we
11 have done that. If you review our Reg Guide, we have
12 a paragraph about it, and you'll see that when you get
13 the Reg Guide, that specifically refers to the -- you
14 know, Reg Guide 1.200, ANS fire PRA standards, so that
15 we can put ourselves and the licensees to a part of
16 convergence.

17 The two remaining issues that we are
18 addressing at the present time are things related to
19 self-approval and cumulative risk. I just listed
20 these for your information. I would request that
21 people not go into a whole lot of detail, because we
22 are still having discussions as to what is the best
23 thing to do. But when we come to you in December, we
24 -- these issues would be pre-addressed.

25 MEMBER APOSTOLAKIS: In December, you will

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1 send us the document. We will have it --

2 MR. WEERAKKODY: We will work very, very
3 hard to give you the revised documents in December.

4 MEMBER APOSTOLAKIS: Good. Do you have
5 anything else?

6 MR. WEERAKKODY: No.

7 MEMBER APOSTOLAKIS: Member, any more
8 comments?

9 MEMBER DENNING: Well, I think I should
10 comment that I think that we are -- you know, the
11 things that we're seeing here are just the things we
12 really did want to see. I mean, obviously, the
13 shutdown PRA -- fire shutdown PRA -- it is really
14 awfully early in the game to be providing definitive
15 guidance on what our expectations would be in shutdown
16 fire PRA.

17 So I -- I do think it's just --
18 ultimately, I think they are going to want to see
19 that, but I do think it's a little bit premature. But
20 certainly the words that we're hearing here and what
21 you're projecting to the industry is much better, I
22 think, than what we saw before.

23 MEMBER APOSTOLAKIS: Any other comments?

24 MEMBER POWERS: Well, we've spoken now
25 about operating events, fires, and a little bit on

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1 shutdown fires. And then there's the other question,
2 which I don't know exactly how to confront, but that's
3 seismically-induced fires. Again, it's painfully
4 obvious what happens in earthquakes. Well,
5 presumably, large concrete robust structures very
6 seldom fail, but very, very often you see fires in
7 those large robust concrete structures.

8 And so the question comes up: what of
9 those systems/situations? And what I worry about is
10 that, again, we -- we only risk-inform that which is
11 easiest to risk-inform, and we're -- we're failing to
12 address where the important issues are, because of the
13 lack of some computer code.

14 DR. GALLUCCI: This is Ray Gallucci. Both
15 the fire PSA standard and NUREG/CR-6850 do address
16 seismic fire interactions.

17 MEMBER POWERS: And have we seen those,
18 Ray?

19 DR. GALLUCCI: You've seen NUREG/CR-6850.
20 That's the risk requant study. I don't think you've
21 seen the fire PSA standard. But it follows -- it's
22 essentially -- it follows pretty much NUREG/CR-6850 on
23 a higher level. There are -- there is a specific
24 element for seismic fire interactions with the
25 supporting requirements for it. And it says it

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1 parallels what's in NUREG/CR-6850, which you have
2 seen.

3 MEMBER APOSTOLAKIS: Well, okay. If it --
4 it's in a NUREG report, what does that mean? The real
5 action is here. Let me ask you this. This fire PRA
6 applies to the power operation, right?

7 DR. GALLUCCI: Yes.

8 MEMBER APOSTOLAKIS: So if the earthquake
9 occurs during power operation, should they have
10 included there seismically-induced fire?

11 MR. WEERAKKODY: Yes. And this is how the
12 connection is made, and then --

13 MEMBER APOSTOLAKIS: Okay.

14 MR. WEERAKKODY: -- in our Reg Guide we
15 refer to Reg Guide 1.200. And one of the appendices
16 of Reg Guide 1.200 is going to be the ANS fire PSA
17 standard. And like Ray mentioned, the ANS fire PSA
18 standard contains the necessary-to-take-a-look-at
19 seismic-induced fires.

20 I do like to make one -- one comment with
21 respect to the shutdown risk and the low-power mode.
22 I think it's not that the staff is not hearing your
23 concern. What our preference is is some of those
24 broader issues be handled under the broader context
25 through the appropriate -- you know, for example, the

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1 low-power and shutdown modes, the fact that we are
2 managing shutdown risk as opposed to low -- you know,
3 developing low-power shutdown, risk assessment is
4 something that is evolving. And on issues like that,
5 we'll develop SPSB and basically follow them.

6 MEMBER APOSTOLAKIS: Okay. Any other
7 comments from the members? From staff? Public?
8 Members of the public?

9 Okay. Well, thank you very much,
10 gentlemen. In fact, I'm very pleased by the way this
11 is going. So I'm looking forward to receiving the
12 document in December, and taking it from there. Thank
13 you very much.

14 Back to you, Mr. Chairman.

15 VICE CHAIRMAN SHACK: Okay. Again, thank
16 you, gentlemen, for an excellent presentation.

17 We're a little bit ahead of schedule
18 again, but don't run off yet, because we would like to
19 take this opportunity to at least have a first reading
20 of some of Mario's letter.

21 We can go off the record for this
22 discussion of the letter.

23 (Whereupon, the proceedings in the
24 foregoing matter went off the record at
25 1:44 p.m. and went back on the record at

1 2:33 p.m.)

2 VICE CHAIRMAN SHACK: We have a real treat
3 ahead of us now -- Davis-Besse Reactor Pressure Vessel
4 Head Integrity Calculations. And Jack will lead us
5 through this.

6 MEMBER SIEBER: Okay. Thank you, Mr.
7 Chairman.

8 I would comment that this issue has been
9 around for a while, and I think most of us were here
10 in 2002 when the cavity in the Davis-Besse reactor
11 vessel head was discovered by the licensee. And a lot
12 of folks like myself speculated, you know, how bad is
13 this really?

14 And a simple-minded way to approach it,
15 like a plant operator would, is to say, "Well, the
16 failure frequency is 1, and, therefore, CDF is totally
17 a function of the reliability of mitigating systems."

18 And you can come up with a number that
19 way, but it's not very satisfying, because everyone,
20 including myself, was curious as to if they had a
21 transient at the plant that would raise reactor
22 pressure to the PORV setpoint, or an ATWS, which goes
23 beyond that, would the head have failed?

24 If nobody did anything and they had enough
25 fuel, how long would they run before it would fail all

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1 by itself? And what is the failure probability,
2 including uncertainties, the year prior to the time of
3 discovery? And these are three basic questions, which
4 the staff and its contractor -- Oak Ridge -- has
5 sought to investigate and answer.

6 And this afternoon's presentation will
7 address that report, and to help us along and get us
8 started I'd like to introduce Alan Hiser to give the
9 staff's introduction. Alan?

10 MR. HISER: Good afternoon. I'm Alan
11 Hiser, Chief of the Component Integrity Section, the
12 Office of Nuclear Regulatory Research. As you
13 mentioned, there have been -- there are several
14 aspects of Davis-Besse that we have looked at, and I
15 think you mentioned several of them.

16 You know, first, looking at the as-found
17 condition and the -- as you know, the margin to
18 failure to that condition. We also looked at analyses
19 to support the ASP analysis, which I -- I think is one
20 of the ways that this presentation came about during
21 the presentation by Gary DeMoss and company in April.

22 In addition, we supported the SDP process.
23 So three sort of distinct sets of calculations.

24 We actually completed this work a little
25 over a year ago, so Dr. Mark Kirk, who will be making

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1 the bulk of the presentation, and I have -- have had
2 to go back and try to refresh our memories on some of
3 the details. So if we're a little bit rusty on some
4 of the facts, you know, please excuse us. But I guess
5 what I would like to do is go ahead and introduce
6 Mark, who will make the presentation on this.

7 MEMBER SIEBER: Well, I can't imagine Mark
8 being rustic.

9 (Laughter.)

10 MR. KIRK: Well, you know, in the presence
11 of boric acid, most things just give way.

12 (Laughter.)

13 I'd like to think I'm austenitic, but
14 maybe not.

15 Anyway, I've also got up here, as a list
16 of co-conspirators, the people that really did the
17 work, which are contractors with the HSST program at
18 Oak Ridge. Those include, of course, Richard Bass,
19 who leads the project; Paul Williams and Sean Yin, who
20 did the bulk -- excuse my voice -- the bulk of the
21 finite element calculations; and then Wally McAfee and
22 Richard were responsible for the burst-test
23 calculations.

24 So the objectives of our analysis, as Alan
25 has already pointed out, were threefold. And I'll go

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1 into each of these in some level of detail.

2 First, we looked at the as-found condition
3 and tried to figure out how much more pressure it
4 would have taken, given the geometric and material
5 conditions on the day of discovery, to have
6 compromised the primary pressure boundary.

7 We did that just because it was a question
8 that many people were interested in, but also it was
9 really the only reality benchmark we had. All that we
10 really knew was that that configuration on that day
11 did not fail. And so we felt it was important, indeed
12 critical, to instilling confidence in our analytical
13 procedures that our analysis should also predict that
14 that geometry on that day under those conditions did
15 not fail.

16 We then did what I've called both a
17 forward-looking and a backward-looking analysis. The
18 forward-looking analysis started with that material
19 condition and geometry and tried to project forward in
20 time based on estimates of corrosion, crack growth
21 rates, in the austenitic stainless steel cladding, and
22 general corrosion rates in the ferritic steel, and
23 tried to project how much longer the cavity might have
24 lasted -- I'm sorry, the cladding might have remained
25 intact under the operating pressure.

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1 And then, we also did what I've called a
2 backward-looking analysis to support Gary DeMoss' ASP
3 calculations, where we tried to postulate what the
4 conditions were a year before February 16, 2002, and
5 that's a requirement of the ASP program, and then
6 calculate forward to get some estimate of the risk of
7 the cladding giving way on the day of discovery.

8 So, again, I'll go into those in that
9 order, but I would like to start with a description of
10 the as-found state, and then I'll talk about our
11 analysis methodology and results.

12 So I don't think I'm going to -- well, I'm
13 certainly not going to show you any pictures that
14 aren't available in the public domain, and I think
15 some of these have been more widely seen than others.
16 On the left-hand side of the screen you have several
17 views of the cavity that was carved out of the reactor
18 pressure vessel head by the boric acid.

19 On the top right, you see a piece labeled
20 "piece M." That's a cross-section through the
21 austenitic stainless steel cladding where the
22 undersurface is the surface that would have been
23 exposed to the pressure of the primary circuit. And
24 the top surface, the undulations in that, are a result
25 of the variable penetration of the weld overlay

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1 process.

2 And then, the green blob-ish looking thing
3 in the lower right-hand corner is the -- I guess the
4 now famous or infamous dental mold that BWXT took of
5 the inside of the cavity. They did it originally
6 under contract to Framatone and FENOC for purposes of
7 just examination.

8 In our effort, we actually digitized that
9 and put it into the finite element model. And I'll
10 show you that.

11 We also contracted separately -- well,
12 through our Oak Ridge contractor -- with FENOC --
13 James Hyres in particular -- I'm sorry, not with
14 FENOC, with BWXT -- the hot cells down in Lynchburg --
15 and Jim Hyres in particular, to perform a more
16 detailed characterization of the flaws in the cladding
17 to support our finite element calculations.

18 And we have reports on that that I believe
19 are available to you. If not, we can certainly make
20 them available. In any event, just a few insights
21 from that analysis.

22 One is this, on the left-hand side, shows
23 a piece of the cladding, and you can see the full
24 cladding thickness, and then the -- the darkened areas
25 are the areas of in-service cracking. So you're

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1 seeing the surface oxides that developed due to the
2 boric acid corrosion.

3 And overall from this we found out that
4 the maximum crack depth was about a tenth of an inch,
5 more like 65 mils on average. And while the surface
6 of the exposed cladding was, in fact, a maze of very
7 shallow cracks, there was one area where the cracks
8 were particularly deeper in between two adjacent weld
9 beads that extended over a crack length of about two
10 inches, where the central two-thirds of an inch had
11 significant depth of the kind shown on the left-hand
12 side and appeared to be more open to the surface, as
13 you can see from the photograph.

14 Also important to our investigation was
15 understanding the crack extension mechanism. The
16 typical microstructure of three -- of -- well, it is
17 a weld metal alloy -- stainless steel is you get a
18 dendritic solidification structure where here the dark
19 areas are the ferrite, the white areas are the
20 austenite.

21 And I would just point out that the
22 presence of ferrite is intentional in the 308
23 stainless steel. It's put there to avoid hot cracking
24 during the welding process. So it's not a mistake;
25 it's supposed to be there. Of course, nobody ever --

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1 it's required to be there, or it's not 308 stainless
2 steel.

3 Of course, it's not designed for exposure
4 to concentrated boric acid, so the concentrated boric
5 acid did to those little islands and pools of ferrite
6 exactly what it did to the rest of the ferritic steel
7 in the RPV head, and it just --

8 MEMBER POWERS: Let me --

9 MR. KIRK: -- took it right out.

10 MEMBER POWERS: Let me understand
11 carefully. You said it's not designed for being in
12 the presence of concentrated boric acid. I mean, it's
13 clearly -- 308 fairly routinely is exposed to boric
14 acid.

15 MR. KIRK: Yes. But not -- not to that
16 level of concentration.

17 MEMBER POWERS: What is the threshold
18 between acceptable and --

19 MR. KIRK: I don't know. And that's
20 certainly beyond my area of expertise. I can get you
21 an answer for that, but --

22 MEMBER POWERS: I mean, do you have -- do
23 you have a sense of it?

24 MR. KIRK: No, I don't.

25 MEMBER POWERS: I mean, ordinary boric

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1 acid is about .1 --

2 MR. KIRK: Well, the only sense I could
3 give you is probably the same one you already have,
4 that at the level of concentration in the primary
5 pressure circuit everything is just fine.

6 MEMBER POWERS: And that's like .1 molar?

7 MR. KIRK: Again, you're outside of my
8 area. I'll defer to anybody who can --

9 VICE CHAIRMAN SHACK: 2,800 ppm boric --
10 boron, I would have to compute that into boric acid,
11 into molar quantities.

12 MEMBER POWERS: .1 mole or something like
13 that?

14 VICE CHAIRMAN SHACK: Probably.

15 MEMBER POWERS: And so it has to be more
16 concentrated than that.

17 MR. KIRK: Yes.

18 MEMBER POWERS: Is a factor of 10
19 sufficient?

20 MR. KIRK: Probably more than that.

21 MEMBER POWERS: So it's essentially boric
22 acid is what --

23 MR. KIRK: Yes, it's a saturated boric
24 acid solution that really causes the problem.

25 MEMBER POWERS: Why would that be?

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1 MEMBER KRESS: Surely it doesn't have to
2 be saturated. It could be some level below that,
3 because there's nothing magic about the saturation
4 level, unless it's a precipitant that does it.

5 VICE CHAIRMAN SHACK: The other thing that
6 probably is also very helpful is typically to have
7 some oxygen available, which you don't have on the
8 other side of the boundary.

9 MEMBER SIEBER: Which is essential.

10 VICE CHAIRMAN SHACK: Well, I don't know
11 that it's essential, but it certainly makes the
12 process a whole lot worse.

13 MEMBER SIEBER: But, I mean, you always
14 have oxidant around.

15 VICE CHAIRMAN SHACK: In the coolant
16 system, you know, it's very, very low levels.

17 MEMBER SIEBER: But it's there.

18 VICE CHAIRMAN SHACK: Yes. I mean, you
19 know, yes, definitely that.

20 MEMBER POWERS: Yes.

21 VICE CHAIRMAN SHACK: If you need PPM and
22 you've got PPB.

23 MR. KIRK: Okay. Well, again, if that's
24 of interest, certainly my colleague Bull Cullen would
25 be much better suited to answer it than me. I can get

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1 that and get it back to you.

2 VICE CHAIRMAN SHACK: Appreciate it.

3 MR. KIRK: Yes. In any event, the bottom
4 figure, which is an optical metallograph, where on the
5 left-hand side is the part of the cladding that was
6 exposed to the boric acid solution in the cavity, and
7 what you see is that the cracks in the stainless steel
8 cladding formed when the concentrated solution
9 preferentially dissolved the ferrite phase. so the
10 cracking is, therefore, in a granular, i.e. between
11 the austenite grains.

12 Now, this slide I think is a particularly
13 important slide, certainly not from a numerical
14 analysis viewpoint, because all you see is pictures,
15 but even more important is this is the expert, this is
16 the metal, this is what was there on February 16,
17 2002. And when we look at it in the scanning electron
18 microscope -- I'll lead you through the pictures.

19 On the upper right-hand side is just a
20 macro photograph where each of those red ticks is --
21 let me refresh my memory -- I think .025 inches each.
22 So the total crack depth there is about a tenth of an
23 inch, and then we've zoomed in on the -- the
24 light/dark interface where the dark part is the crack
25 that developed in service.

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1 And what we see when we go to the highest
2 magnification of 500x is that on the dark side you get
3 the intergranular cracking that's characteristic of
4 the boric acid attack, but you don't -- and this is
5 the significant part -- in the service darkened area,
6 you don't see any evidence whatsoever of micro-void
7 coalescence that would indicate the ductile overload
8 type of failure that we understand on the basis of our
9 burst test, which I'll explain in a minute, is the way
10 that the cladding would have ruptured had it ruptured.

11 So the point to be taken away from this
12 slide is that the forensic evidence that's clearly
13 evident in the cracks, in the stainless steel
14 cladding, show that while the cladding did appear to
15 have been deformed by the service loads, there is
16 absolutely no evidence of ductile crack initiation.

17 So there is no indication from the
18 forensic evidence that this cladding could in any way
19 be characterized as ready to go. And that's, again,
20 an important point to take away from a failure
21 analysis viewpoint.

22 There's also an important point to take
23 away from the viewpoint of benchmarking our finite
24 element analysis, in that not only should our finite
25 element analysis of this geometry under this material

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1 condition not predict failure, but it should also
2 definitely not predict that the applied J-integral
3 values exceed the J_{1C}, which would mean if that
4 happened that would mean you should be seeing ductile
5 growth in the service darkened regions.

6 So that's it for our summary in this
7 presentation of the forensic exams. I'll now go on to
8 talking about our methodology for integrity assessment
9 of the vessel head in the as-found state. And this is
10 just a cartoonish-type schematic showing you the
11 various inputs that were needed.

12 We, of course, characterized the as-found
13 condition, and we talked a little bit about that. We
14 calibrated our failure model using large-scale tests,
15 or I should actually say validated it. That, then,
16 both served as inputs to a finite element model,
17 which, along with material properties, allowed us to
18 assess the structural condition of the cavity.

19 So for input information to that analysis,
20 and a bit more detail, we needed to know, of course,
21 the geometric configuration of the cavity, and the
22 crack size and distribution. And while I've just gone
23 into some level of detail showing you that on the
24 preceding slides, it should be appreciated that when
25 the initial analyses were being conducted in the

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1 heated days just after the cavity was discovered, most
2 of that information was not available.

3 In fact, it didn't all become available
4 until sometime in the spring of 2004 after which our
5 analyses were finalized. So that's why if you tuned
6 in to the results of our analysis at various points in
7 time, you'll see somewhat differing results, different
8 conceptions of what the factor of safety against burst
9 was, and how much longer the cavity could have lasted,
10 because we've been continuously refining our models.

11 So we needed that information to do a
12 credible analysis. We also needed information on the
13 cladding strength and fracture toughness properties,
14 and we needed to perform -- we decided to perform our
15 burst test experiments to confirm our ideas about how
16 the cladding would have failed, had it failed, and to
17 benchmark our predictions.

18 So in terms of cladding strength, here
19 you've got a bunch of true stress/true strain curves
20 that we collected from the literature, and overlaid in
21 the middle of that you see BWXT specimens, two of
22 them, and those are specimens -- little tiny tinsels
23 that were pulled directly from the Davis-Besse
24 cladding material. So you can see that, from a
25 stress-strain point of view, the material is entirely

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1 typical of 308 stainless steel.

2 And I'll just point out in passing that we
3 used that information, then, to construct probability
4 distributions, that we then used in our Monte Carlo
5 analysis when we were looking at predicting failure
6 probabilities.

7 Similarly, we needed to know the ductile
8 fracture toughness of the cladding material at the
9 surface temperature. The results of tests that we
10 performed -- if I can get a pointer here somewhere.
11 Oops, sorry.

12 The results of tests that we performed on
13 fracture tuft and specimens removed from the Davis-
14 Besse cladding are shown here. And when making all
15 your comparisons at the same test temperature, you see
16 that, again, the Davis-Besse cladding is fairly
17 typical of a 308 stainless steel. Sometimes we can
18 find properties that are not as tough. Sometimes we
19 can find properties that are more tough.

20 Again, there is nothing particularly
21 atypical about this particular material from a
22 strength and toughness point of view.

23 As I have mentioned several times, we did
24 a series of burst tests at the Oak Ridge National
25 Laboratory, and here you're looking at sort of the

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1 meat of the burst test where we took a plate of an RPV
2 cylinder that was made for plant service but never
3 installed in plant service. This is what's frequently
4 referred to as the PVRUF material.

5 There was a six-inch thick reactor
6 pressure vessel steel plate that had been clad using
7 standard industry practice. We then -- our colleagues
8 at Oak Ridge then machined a six-inch diameter hole
9 six inches deep into that plate of steel, leaving only
10 the cladding material.

11 Some of those -- so we had a six-inch
12 burst disk, which was meant to fairly closely
13 represent the same unbacked area of the cladding that
14 was in Davis-Besse. We also did tests at a number of
15 fall depths, with the intention of both bracketing the
16 fall depths that we observed in Davis-Besse, which was
17 about a tenth of an inch out of a quarter-inch
18 thickness of cladding, and also by performing tests --
19 parametric and fall depth -- we were able to examine
20 the effect of fall depth on the failure mode.

21 I would point out something here that, you
22 know, we had some trouble getting away from is to
23 dissuade anybody from the notion that this test is
24 intended in any way to be a one-for-one model or
25 representation of Davis-Besse. It isn't. Clearly, it

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1 isn't. The shape isn't right, and there is no
2 corrosive environment, and it's not done at 600
3 degrees Fahrenheit.

4 So there are quite a few things that are
5 different, but what we wanted to do here was to -- to
6 replicate fall depth and unbacked area in an effort to
7 get something close that we could benchmark a model
8 on, and then use the model to capture the much more
9 complex geometric and environmental variables that
10 were difficult to test.

11 So the objective of performing these tests
12 was to either validate or refute the opinion
13 ourselves, and I think most people that looked at it,
14 that the cladding would tear by -- would fail -- I'm
15 sorry -- by either a ductile tearing or an overload
16 mechanism, and also to assess the accuracy or
17 conservatisms in our predictive fracture mechanics
18 models.

19 So there is the picture of what the
20 specimen looked like before the test. After the test,
21 if you had a crack of fairly substantial depth -- and
22 by "substantial" I mean two-tenths of the way into the
23 cladding thickness or more, and that's certainly the
24 condition that existed at Davis-Besse on 2/16/02 after
25 the test. And this is now the six-inch test section

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1 that has been cut out of the rest of the reactor
2 vessel steep.

3 You've just got a nice bulging out.
4 Ultimately, the crack -- and this is the -- this is
5 the pressurized surface, this is the non-pressurized
6 surface. Ultimately, the crack tore through, released
7 the pressure, and the test was over. We got a
8 fundamentally different response from our specimens
9 when there was either a very shallow crack, something
10 like 10 or 15 percent of the way through the
11 thickness, or no crack at all.

12 In that case, while certainly being less
13 cracked indicates -- and it's, in fact, true, that the
14 test specimen or the structure, if you want to call it
15 that, could withstand a higher load, when the specimen
16 or structure actually failed, the failure was quite
17 catastrophic. And what you're seeing is that the
18 central disk was completely ripped out of the test
19 fixture, and, in fact, cost us several thousand
20 dollars in lost instrumentation until we decided to
21 stop performing tests like that with instrumentation.

22 So to summarize the results and compare
23 them with our predictions, on this slide the graph --
24 the blue dots on the graph are the results of the
25 test, and the results are presented as the critical

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1 pressure divided by the cladding thickness plotted
2 versus the crack depth, normalized again by the
3 cladding thickness.

4 The set of sweeping curves show you the
5 mean prediction and confidence bands on failure when
6 failure is by initiation of stable duct of tearing,
7 whereas the upper lines show you the -- I guess it's
8 better to say the median prediction and the
9 uncertainty bounds when failure occurs by overload of
10 plastic collapse.

11 And what you see in the test data is a
12 transition between those two failure modes, where if
13 you have either no flaw or fairly shallow flaw the
14 overload plastic collapse type of failure dominates.
15 And while you do get lower failure load -- or, I'm
16 sorry, higher failure loads, failure pressures, you
17 tend to blow out the entire unbacked area, so you get
18 a much larger break in the -- if it were the pressure
19 circuit, in the pressure circuit.

20 Whereas, when you get the stable tearing
21 type of failure, you fail obviously at much lower
22 pressures, but the size of the opening is expected to
23 be considerably less.

24 MEMBER POWERS: All gas pressurized
25 systems?

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1 MR. KIRK: Alan, do you remember? I'm --
2 yes, it's gas.

3 MEMBER POWERS: If you had done the test
4 with cracks, and hydrostatically loaded it, would it
5 have been -- just left that little fine crack you
6 showed, or would it have ripped open --

7 MR. KIRK: Obviously, there would be a
8 greater tendency to rip a larger hole.

9 MEMBER POWERS: So is this --

10 MR. KIRK: But there --

11 MEMBER POWERS: -- without a difference
12 here?

13 MR. KIRK: I think, you know,
14 qualitatively it's going to go that way.
15 Quantitatively, we just -- we haven't covered that in
16 our analysis.

17 MR. HISER: I'm sorry. What was the
18 context of the question again?

19 MEMBER POWERS: Well, the distinction has
20 been made here that with a crack you get this -- and
21 it vents the pressure out, because it's gas-loaded.
22 Whereas with no crack, it blows the entire disk out.
23 What I ask is, gee, if you hydrostatically loaded it
24 instead, wouldn't the post-test examination have been
25 about the same?

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1 MR. KIRK: That's a good question. The
2 only thing I could add in is that the -- the
3 calculations that we ran to actually do the integrity
4 assessment effectively did keep the pressure on and
5 calculated the stability of the crack once it tore
6 through. So while that feature, indeed, as you've
7 pointed out correctly, is not well captured in our --
8 in our test, it is well captured in the analytical
9 model.

10 MEMBER POWERS: I'm just trying to
11 understand --

12 MR. KIRK: Yes.

13 MEMBER POWERS: -- what I'm supposed to do
14 with this information, and it strikes me I'm not going
15 to do anything with it. When it overpressurizes, it
16 busts big time. And there's -- I mean, that's the
17 message I get.

18 MR. HISER: Well, but I think there's a
19 couple messages. I think the one message is that, you
20 know, it's a race to failure. If you had a -- a
21 static load condition, you know, constant pressure,
22 and the cavity is growing, the cracks are growing
23 deeper, you know, one of them is eventually going to
24 get to a failure condition. And which one gets there
25 faster is the one that would probably determine

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1 whether you blow out the cavity or you end up with a
2 leak.

3 MEMBER POWERS: No, I'm not sure I'd --
4 I'm not sure that's the part I bought.

5 VICE CHAIRMAN SHACK: You know, you get a
6 fishmouth if you -- you know, if you had a load, you
7 get a fishmouth rather than that little tiny crack.
8 I mean, you presumably did predict deformations.

9 MR. KIRK: Yes. But as you know, trying
10 to go to actually that predictive level, but I would
11 emphasize is that when we did the calculations in the
12 forward- and backward-looking analysis, once the crack
13 tore through we were then able to assess stability of
14 the torn-through crack and determine whether it would
15 continue to rip or rip out.

16 MR. HISER: And I think in general the
17 calculations -- when you first get the leak, the crack
18 doesn't suddenly go unstable.

19 MR. KIRK: No, it doesn't.

20 MR. HISER: And so you would ultimately --
21 you know, there's going to come a point where you're
22 going to detect leakage through leak detection
23 methods.

24 MEMBER POWERS: Every place except perhaps
25 at Davis-Besse, I would --

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1 MR. HISER: Well, but I think that -- that
2 really is why this part of the calculation is
3 important, because, you know, whether you blow out an
4 area that's larger and you get the equivalent LOCA
5 from that, or you get a leakage through a slit sort of
6 mechanism that, you know, maybe the crack is growing,
7 but it still maintains stability because the material
8 has fairly high tolerance.

9 MEMBER POWERS: Do you calculate stream
10 erosion when you calculate these crack stabilities?

11 MR. KIRK: No.

12 MEMBER POWERS: Stream erosion, it seems
13 to me, would at some point dominate here.

14 VICE CHAIRMAN SHACK: Well, you'd
15 certainly be above your tech spec limit.

16 (Laughter.)

17 And hopefully be shutting down pretty
18 fast.

19 MEMBER SIEBER: Hopefully.

20 MEMBER POWERS: Well, it still would take
21 the load off, Bill.

22 MR. KIRK: Moving on, further looking at
23 the geometric inputs, the finite element model of the
24 as-found state, we've shown here in detail, more in
25 our reports, how the dental mold was used to get an

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1 accurate representation of both the footprint of the
2 wastage area as well as the three-dimensional
3 geometric shape.

4 And that, then, on the right-hand side, of
5 course, you see the mold; on the left-hand side, a
6 graphical representation of the mathematical model of
7 the mold that was then used to establish geometry for
8 the finite element model.

9 We also incorporated into the finite
10 element model the average periodicity of the welding
11 causes the crenulations on the inside surface of the
12 cavity. And the lower figure just illustrates that we
13 located the crack in our cavity model in the same
14 place that it was found in the service condition.

15 Again, another view of -- showing you the
16 details of the finite element model, to point out that
17 just for purposes of actually getting the calculation
18 done, you know, sometime at least before the end of my
19 career, we have to take a substructure approach where
20 we started off by modeling the whole head without the
21 CRDM penetrations.

22 We then carved out a little pie-shaped
23 sector, applied the boundary conditions on the pie-
24 shaped sector determined from the bigger model,
25 modeled the effect of the CRDM penetrations, and the

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1 hole and the cladding, and put the crack in that
2 model.

3 MEMBER SIEBER: I would presume the shape
4 of the wall in the calculation has no effect on -- on
5 burst strength or stress or characteristics.

6 MR. KIRK: To be honest, probably not.
7 However, having gone through multiple iterations with
8 less elegant models and not being able to predict with
9 any degree of believability the fact that this
10 geometry had not failed, we eventually just pulled out
11 all the stops and said, "Okay. Let's model everything
12 we possibly can." But I would agree. The only --

13 MEMBER SIEBER: I think the footprint is
14 important.

15 MR. KIRK: The footprint is certainly
16 important. The only thing --

17 MEMBER SIEBER: But the wall shape is not.

18 MR. KIRK: The only thing -- the only
19 feature that I think was probably important to
20 capture, but, again, we didn't do a sensitivity study
21 to show this -- is this -- this nose or little area of
22 overhang here, where you've got material here that's
23 only backed by a very small thickness of the ferritic
24 material.

25 MEMBER SIEBER: Okay.

1 MR. KIRK: So I think probably, you know,
2 of all of the complex features of that shape, that's
3 the one that was important. But to get that we
4 modeled the whole thing.

5 Okay. So going on, so our as-found
6 analysis based on a geometric finite element model to
7 estimate stresses, the actual properties of the Davis-
8 Besse material for the cladding for strength, the
9 actual Davis-Besse properties for the cladding
10 fracture toughness, and because the actual condition
11 was actually a network of interlinking cracks, to make
12 the model tractable we idealized that into three
13 different representations of that network of cracks.

14 I'm just going to focus on one that we
15 called our bounding model, where we bounded the depth
16 of that network of cracks at a tenth of an inch in the
17 length, at two-tenths of an inch. So the results of
18 the as-found analysis are shown here. I'd like to
19 focus your attention on the graph.

20 The vertical axis is J applied or the
21 driving - the applied driving force to fracture that
22 occurs as a consequence of the pressure loading. The
23 three different colored curves represent our three
24 different flaw models, and, again, I'll just focus
25 attention on the -- what we've called the conservative

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1 bounding model or flaw number 3 that's shown in blue.

2 The three horizontal lines represent the
3 range of fracture toughness characteristic of the
4 Davis-Besse material at the 95th median and 5th
5 percentiles. And, to me, the takeaway point from this
6 presentation is that the operating pressure we're
7 nowhere near the 5th percentile J1C. And even at the
8 setpoint pressure we're still below the 5th percentile
9 J1C.

10 So our prediction would have -- you know,
11 if somebody asked us to predict this, which I guess
12 they did, is that failure didn't occur, and, moreover,
13 hey --

14 (Laughter.)

15 -- that was only a few million dollars and
16 several years later. And the ductile crack initiation
17 didn't occur, and, in fact, that's what occurred in
18 service.

19 The other I think heartening thing to take
20 away from this is that the difference between the
21 operating pressure and the relief valve setpoint
22 pressure was not adequate even getting a bounding flaw
23 characterization, and even given a bounding fracture
24 toughness characterization to compromise the integrity
25 of the cladding.

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1 MEMBER SIEBER: But that's about a 10
2 percent increase in pressure.

3 MR. KIRK: That's right.

4 MEMBER SIEBER: Yes.

5 MR. KIRK: Yes. So at least to me the
6 takeaway from this is that the -- in reality,
7 obviously, the probability on failure of date of
8 discovery was zero. But based on this analysis,
9 assuming the set valves work -- and I'll leave the
10 probability of that to others that know better -- is
11 exceedingly low.

12 Okay. So now working on to our forward-
13 and backward-looking analysis, basically the same
14 analysis/methodology. We need a few more inputs, and
15 we also needed to develop from our very detailed
16 three-dimensional finite element model a much more
17 simplified model just to enable the forward- and
18 backward-looking calculations.

19 And so an in-going assumption to our
20 analysis is that the -- I shouldn't say the complex
21 cavity shape -- the complex footprint shape can be
22 modeled as a circle. And I provided -- and at first
23 blush that looks like an awful gross approximation.
24 I'll give you two scientific reasons and one practical
25 reason why you should maybe let me get away with that.

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1 The scientific reasons is that for failure
2 by plastic collapse the total unbacked cladding area
3 is a much, much more important parameter than the --
4 than the unbacked area shape. And as evidence of
5 that, I provide you the graph shown here, where the
6 downward-sweeping curve is, in fact, a closed form
7 plasticity solution due to Chakrbady and Alexander,
8 published in 1970, of exactly this geometry.

9 And then, we performed a number of
10 different finite element analyses, both where we took
11 the sort of boot-shaped footprint and expanded itself
12 similarly, and we looked at different ellipsoidal
13 growth patterns. And for all intents and purposes,
14 given the other approximations in the analysis, all
15 the points were pretty darn close to the theoretical
16 circular growth pattern.

17 So, again, for the plastic overload type
18 of failure, the shape really just doesn't matter. For
19 failure by ductile tearing, the circular consumption
20 -- I'm sorry, the circular assumption is indeed
21 conservative, because when you put the crack in the
22 middle of the disk, as we did, you know, that the
23 crack, because of the geometry, has to be oriented
24 perpendicular to the principal stresses.

25 Whereas, we know that the crack in Davis-

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1 Besse formed preferentially due to the -- the
2 metallurgy of the ferrite stainless steel and the
3 boric acid in the cavity, and that turned out not to
4 be oriented perpendicular to the applied principal
5 stresses.

6 So when we assess the crack in the cavity
7 as a crack in the circle were, in fact, overestimating
8 the driving force to fracture. So those are my -- my
9 scientific reasons why this is a reasonable thing to
10 do. The somewhat non-scientific reason is we just
11 don't know anything better to do.

12 The corrosion experts were unwilling to be
13 -- and I think justifiably so -- be boxed into a
14 corner to provide any kind of a quantitative model by
15 which either the cavity developed to the shape it was
16 or would have proceeded from there on. So given that
17 lack of modeling information, a circle is about as
18 good as anything else.

19 MEMBER POWERS: I just can't resist. You
20 would get an A+ in our quality review for
21 justification of assumptions here.

22 MR. KIRK: Thank you.

23 MEMBER POWERS: In the first place, circle
24 was -- I mean, a cylinder looked like a pretty good
25 approximation to me to begin with, and you've

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1 convinced me that it's an excellent approximation.

2 MR. KIRK: Well, remember, I've had three
3 years to think about this.

4 MEMBER POWERS: And my third thing is I'd
5 be willing to take on trying to calculate based on --
6 on corrosion, what the shape of the cavity is.

7 MEMBER SIEBER: Did you say the score
8 would be .8?

9 (Laughter.)

10 MR. KIRK: 42. I think the answer is 42.

11 MEMBER POWERS: I believe you'd get a
12 solid 5 on this one.

13 MEMBER RANSOM: Well, I know where the as-
14 found model for 2004 and the as-found model for
15 2002 --

16 MR. KIRK: Hang on. I'm refreshing. Yes.
17 As I said, our state of knowledge regarding what the
18 footprint of the cavity was and what its shape was
19 evolved significantly over time. The original as-
20 found model, September 2002, was I think based on
21 poroscopic measurements and somebody sticking a ruler
22 down into it, and sketches made by inspectors.

23 By the time we got to 2004, we had the --
24 the green pukish-looking dental mold, so we had a much
25 more accurate representation. So that's just

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1 difference in state of knowledge.

2 Okay. So the input information to these
3 calculations, we needed to have -- since we're doing
4 a probabilistic analysis, our inputs need to be
5 statistically distributed. So we needed to have
6 statistical representations of toughness and strength,
7 which you've already discussed. Some things we had to
8 base on engineering judgments, and I'll talk a little
9 bit about that -- our rules for LOCA binning and our
10 statistical fitting of data.

11 Other things were based on what I've
12 called expert opinions benchmarked to data, and that
13 had to do with the general corrosion properties of the
14 ferritic RPV steel and the corrosion crack growth
15 properties of the austenitic stainless steel cladding.

16 It's certainly not to say that data
17 doesn't exist -- in fact, ample data doesn't exit --
18 for both of those phenomena. You could go into the
19 literature and find lots and lots of it.

20 The difficulty was, and where we relied on
21 three internal people with expertise in this area to
22 help guide us, is nobody was ever really sure what the
23 thermal and acidic conditions were in the cavity
24 itself. One of my colleagues referred to that as
25 something like sheer conjecture.

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1 So, anyway, we asked three people to make
2 a sheer conjecture on what that was, and that led them
3 to sometimes differing/sometimes similar views as to
4 what the general corrosion and the stress corrosion
5 properties of the ferritic and austenitic materials
6 was respectively.

7 MEMBER POWERS: If they were looking at
8 general corrosion for the ferritic material, they must
9 have had some estimate of the stability of ferrous and
10 ferrite borates in solution. Yes?

11 MR. KIRK: Presumably, yes.

12 MEMBER POWERS: Do you know what they
13 used?

14 MR. KIRK: I have no idea.

15 MEMBER POWERS: Because, I mean, I know of
16 exactly one report in the literature on the stability
17 of the borate complexes of iron in solution.

18 VICE CHAIRMAN SHACK: Well, I mean, I
19 think these were measured from general just corrosion
20 tests of ferritic steel. I mean, they had the
21 corrosion. Given the temperature and a boric acid
22 concentration, as Mark said, we sort of know the
23 corrosion rate. What we don't really know is what the
24 temperature and the concentration is in the cavity.

25 MEMBER SIEBER: Well, it's always

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1 changing, too.

2 VICE CHAIRMAN SHACK: Well, yes.

3 That's --

4 MEMBER SIEBER: And so is the corrosion
5 rate.

6 VICE CHAIRMAN SHACK: Yes.

7 MEMBER SIEBER: And that's what makes the
8 problem difficult is that you have a constantly-
9 evolving situation.

10 MEMBER POWERS: Yes. But it doesn't hold
11 very much.

12 MR. KIRK: Maybe you need more optimistic
13 experts.

14 MEMBER POWERS: We're always optimistic.

15 MR. HISER: If this is even a parameter,
16 we don't even know the end state what it was that they
17 have discovered, because it -- it wasn't sampled. So
18 it's --

19 MEMBER KRESS: Yes. There is one class of
20 opinions that says that that cavity had gone as far as
21 it's ever going to go.

22 MEMBER SIEBER: I had heard that. That's
23 conjecture, though.

24 MR. KIRK: I don't think we had any
25 experts that were that optimistic.

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1 MEMBER KRESS: Well, the reasoning had to
2 do with the size of the -- the opening to the top that
3 would finally relieve the pressure in there, and
4 thereby relieve the concentration and --

5 MR. KIRK: Right.

6 MEMBER KRESS: -- boil away the solution
7 that --

8 MR. KIRK: I was just looking at that. I
9 skipped add and looked at their inputs. Nobody
10 predicted a zero effective cavity wastage rate. So
11 nobody was that optimistic about the situation.

12 MEMBER KRESS: That's really optimistic.

13 MR. KIRK: Well, yes.

14 MEMBER KRESS: You wouldn't have any --

15 MEMBER POWERS: But there are zeroes and
16 zeroes here, and I can't --

17 MR. KIRK: The problem is you're working
18 with --

19 MEMBER POWERS: Yes. Your corrosion rate
20 can always be finite, but it can be so minuscule that
21 it's essentially unmeasurable.

22 MR. KIRK: Well, I don't think they were
23 actually predicting zero.

24 MEMBER KRESS: All right.

25 MEMBER RANSOM: Did the wastage occurred

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1 from the outside and -- in other words, the boric acid
2 concentrate on the outside surface and then go down
3 through the --

4 MR. KIRK: I believe that's one of the
5 models.

6 MR. HISER: Yes. I think there's a lot of
7 conjecture on that as well, whether it ate, you know,
8 down at the -- at the -- near the clad, base metal
9 interface, and then that grew up, or, you know, the
10 concentration flowed up to the surface and then it ate
11 down. I mean --

12 MEMBER SIEBER: Who knows?

13 MR. HISER: Yes, it's -- all we know is at
14 one point in time everything was intact.
15 February 16th it looked that way, and we don't have
16 any data points in between, unfortunately.

17 MEMBER RANSOM: It would seem like the
18 evidence would favor from the outside, because
19 otherwise the concentration would be no different than
20 the concentration on the interior of the reactor
21 vessel, as far as the boric acid concentration.

22 MR. HISER: Well, except you get boron off
23 of the water.

24 MEMBER RANSOM: There has to be a vent or
25 something, though, doesn't there?

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1 MR. HISER: Well, there's an annulus
2 between the CR --

3 MEMBER KRESS: There's a place for the
4 steam to go out.

5 VICE CHAIRMAN SHACK: There's a gap
6 between the -- you know, if you get the crack through
7 the nozzle, then there's a gap for the steam to
8 escape.

9 MEMBER KRESS: It depends on the size of
10 that gap as to whether it concentrates it or
11 deconcentrates it.

12 VICE CHAIRMAN SHACK: Yes.

13 MEMBER SIEBER: And it's not concentric.

14 VICE CHAIRMAN SHACK: Well, it will get
15 larger.

16 MEMBER KRESS: It will get larger. And
17 eventually it'll reach a state where it boils this
18 stuff away --

19 MEMBER SIEBER: Yes.

20 MEMBER KRESS: -- faster than a crack can
21 put it in.

22 MEMBER SIEBER: Right. It's just blown.

23 VICE CHAIRMAN SHACK: But we have plenty
24 of cracks where the wastage did not occur.

25 MR. KIRK: Sorry?

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1 VICE CHAIRMAN SHACK: We have plenty of
2 cracks where the wastage --

3 MR. KIRK: Oh, yes. Absolutely.

4 VICE CHAIRMAN SHACK: So, you know, the
5 exact conditions that produce minimal wastage and the
6 conditions that produce --

7 MR. HISER: Actually, I would maybe
8 caution a little bit on that. There was another
9 nozzle at Davis-Besse that had some incipient wastage
10 down near the clad base metal interface. I'm not sure
11 how many other plants did sufficient examination to be
12 able to detect anything like this.

13 MEMBER POWERS: All we're doing is
14 confirming that metallurgy is not yet a precise
15 science.

16 VICE CHAIRMAN SHACK: No. If you've got
17 essentially your milligram of boric acid on top, it
18 says that not a whole lot came through. I mean, you
19 know, most of those other amounts are associated with
20 like one gallon of total leakage. Well, you know, the
21 amount associated with the leakage here is much
22 larger.

23 MEMBER SIEBER: I think the statement you
24 made needs some expansion. The examinations may not
25 have been sufficient to determine that a cavity was

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1 forming, but they were sufficient to determine whether
2 there was a crack or not.

3 MR. HISER: Yes, that's correct. And
4 that's where the examination is focused on.

5 MEMBER SIEBER: So even though a cavity
6 might have begun to exist, a repair to a place that
7 stopped further progression.

8 MR. HISER: Right.

9 MEMBER SIEBER: Okay. You know, just to
10 leave that hanging, one would think, well, it --
11 there's cavities forming in half the plants, and
12 that's not true.

13 MR. HISER: No.

14 MR. KIRK: Okay. And then, the last
15 category of input information was -- I've also said,
16 based on expert opinion -- and I would say somewhat
17 greater level of conjecture than was the previous
18 bullet, and those are the conditions on the -- of the
19 crack depth in the austenitic stainless steel cladding
20 of the cavity size one year before the situation was
21 discovered.

22 Obviously, the individuals we asked needed
23 the same sort of basic input information, but then
24 they had to, in their minds, back everything up a
25 year.

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1 I should point out -- and I'm perhaps
2 getting a little bit ahead of myself -- that using the
3 information provided by this group of three
4 individuals, and then fitting statistically, we
5 performed our calculations from time of discovery
6 minus a year up to time of discovery. And at least on
7 average they weren't that far off.

8 The crack depths and the exposed area of
9 cladding that we were predicting at time of discovery
10 did not deviate by that much, again on average, from
11 the conditions that were actually discovered. So what
12 the group did on whole, on average, worked out pretty
13 good.

14 The engineering judgments that were made,
15 which I guess is somewhat more guidant than
16 assumptions, in my view, had to do with the local
17 binning rules -- the LOCA binning rules, I apologize
18 -- and the statistical fitting of data. LOCAs were
19 categorized as being small if they produced a break in
20 the primary pressure circuit up to three and a half
21 inches in diameter.

22 And I believe the three and a half inch
23 cutoff was based on what the makeup systems can
24 replace. Medium was 3-1/2 to 4.8, and large is
25 greater than 4.8.

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1 We then had what we called conservative
2 best estimate and less conservative LOCA binning
3 rules, which, again, are detailed in the report. I'd
4 just point out that the conservative model would
5 equate through clad cracking -- in other words, where
6 the crack tip penetrates the cladding layer as
7 complete failure. That was what our conservative
8 models would have given us.

9 Whereas the best estimate model start --
10 took that and then calculated the stability of the
11 through clad crack under the pressurized conditions,
12 and saw if it would tear stably or just let go.

13 MEMBER KRESS: I'm just curious, what's
14 the basis of the 4.8 inch?

15 MR. KIRK: I apologize, but I -- I don't
16 know the answer to that question. I wasn't involved
17 in that.

18 Gary, do you -- I'm getting no. I can
19 find that out for you.

20 MEMBER KRESS: I was just curious.

21 MR. KIRK: Yes, because it's certainly --
22 it's certainly -- I apologize. I wasn't involved with
23 the project in the middle. I got it on both ends, and
24 that happened in the middle. But I can find that out
25 for you.

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1 MEMBER KRESS: It probably has something
2 to do with how fast the primary system depressurizes.

3 MR. KIRK: And then, judgments had to be
4 made regarding how we fit statistical distributions to
5 our judgment information, and that's illustrated on
6 the following slides.

7 So this table is just the input
8 information that we got from our subject matter
9 experts on our four variables -- those being the
10 cavity radius at time of discovery minus one year, the
11 cavity wastage rate or the general corrosion rate of
12 the ferritic steel, the fall initiation time relative
13 to the time of discovery, how long the falls had been
14 in the cladding, and then also the effective flaw
15 growth rate just put these up for information and to
16 illustrate that the inputs given us by the experts
17 tend to span a fairly wide range, as you might expect,
18 given the uncertainties that they had regarding the
19 environment inside the cavity.

20 MEMBER POWERS: You mention frequently the
21 experts. Do you ever reveal who the experts are?

22 MR. KIRK: Alan, should I reveal who
23 candidate 1, 2, and --

24 MR. HISER: No. These are three staff
25 members that -- that have --

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1 MEMBER POWERS: Three guys you grabbed out
2 of the lunch room and --

3 MR. KIRK: Who have --

4 MEMBER POWERS: -- kicking and screaming.

5 MR. KIRK: -- who have far more expertise
6 in the corrosion area than either Alan or I.

7 MR. HISER: We don't even say guys,
8 because you make assumptions there.

9 MEMBER POWERS: Kicking and screaming. I
10 am informed reliably by the current Merriam-Webster
11 dictionary that "guy" is non-sexual. It is uni-sexual
12 now.

13 MEMBER SIEBER: It is?

14 MEMBER KRESS: Yes.

15 MR. KIRK: Now, where I went to school for
16 my bachelor's, which was Virginia Tech, we just say
17 y'all. And when I worked in Pittsburgh, we just said
18 you'ns, and those are also asexual and also not
19 understandable to people that grew up outside of --

20 MEMBER POWERS: Tom understands it
21 perfectly.

22 MEMBER KRESS: I understand y'all, and
23 you'ns, too.

24 MEMBER POWERS: No. You understand y'all.
25 You don't understand you all. You understand y'all.

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1 MEMBER KRESS: But that makes you a Hokie,
2 right?

3 MR. KIRK: That's right.

4 MEMBER KRESS: I don't know what that is.

5 MR. KIRK: And hopefully none of that just
6 got into the minutes, or I'm going to be asked to
7 spell it.

8 (Laughter.)

9 Okay. So here we have the probability
10 density functions that we fit to both the cavity
11 growth rate and the crack growth rate. Oh, that's the
12 old one, never mind. One is the probability density;
13 one is the cumulative probability. I apologize for
14 the difference.

15 But, again, just to point out the cavity
16 growth rate, we were fitting values that ranged from
17 almost nothing per year to up to seven inches per
18 year, and the statistical distributions cover that
19 range. And the crack growth rate in the cladding all
20 the way from almost nothing to a tenth of an inch per
21 month, and then a tenth of an inch per month, given
22 that you've only got quarter-inch cladding, it just
23 doesn't take too long to get through.

24 With only three data points, it isn't
25 surprising to note that you can fit pretty much

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1 anything through there, and your best-fit statistics
2 don't tell you an awful lot.

3 So we took some standard density functions
4 and then categorized them as being either best
5 estimate meaning somewhere in the middle, more
6 conservative meaning tending towards higher values,
7 and less conservative meaning tending towards low
8 values, and then we ran a whole bunch of cases for our
9 Monte Carlo analysis to try to get a sense of the
10 effects of model uncertainty on what we will
11 subsequently label our best estimate, or perhaps best
12 guess values. And here you go.

13 So these are the results of the -- of what
14 I've called the forward-looking analysis where we
15 start with the known as-found state as certain, and
16 then we project forward in time. So on the -- on the
17 left-hand side of your screen you've got the breakdown
18 with LOCA size.

19 Obviously, you've got no failure
20 probability up to the day of discovery, and then the
21 failure probabilities start to kick up, where the red
22 curve is the total LOCA probability, blue is small
23 break LOCA, brown medium break, and green large break.

24 And a thing to point out here is that the
25 small break LOCA dominates, and that's a direct

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1 consequence of the fact that we know from our forensic
2 investigations that the cracks were already a tenth of
3 an inch through a quarter of an inch of the stainless
4 steel cladding.

5 In this case, the deeper cracks actually
6 tend to reduce the consequence of the failure, because
7 even though they claim the failure, had it occurred,
8 would have occurred sooner, there is less energy in
9 the system and, therefore, less likely to blow a big
10 hole in it.

11 Excuse me. I'm losing my voice.

12 On the right-hand side, now looking at
13 just total LOCA probabilities, you see the effect of
14 our three different flaw size idealizations. And the
15 results that we've been, you know, talking about are
16 based on our enveloping flaw characterization, which
17 is shown by the -- by the upper curve.

18 So, again, based on the bounding flaw
19 model, which is flaw 3 -- and I should note that that,
20 while ASME doesn't give practices for enveloping such
21 flaws, they do give interacting flaw practices, and
22 basically we drew a big oval around all of them.

23 Based on the bounding flaw model, our
24 model predicts that there was between 2 and 22 months
25 of operation beyond February 16, 2002, that could have

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1 taken place before the cladding was compromised where
2 the best estimate value, meaning the median value, is
3 five months.

4 Obviously, a pretty wide range there
5 reflecting the uncertainties in projecting this
6 forward based on unknown environmental conditions.
7 But then the bottom point I think is a more certain
8 result because of what we do know about the cracks and
9 the cladding, and that is had a failure occurred it's
10 very much more likely to have been a small break LOCA
11 than a larger break.

12 MEMBER DENNING: Now, that's not
13 necessarily a good thing, right? I mean, as far as
14 conditional core damage and the -- and knowledge about
15 the systems in that plant, it's possible to -- have
16 you looked at -- when you look now and you add on
17 conditional probability core melt, are you better or
18 worse to have a small break LOCA or a large break
19 LOCA?

20 MR. KIRK: I'm going to have Gary talk to
21 that.

22 MR. DeMOSS: I'm Gary DeMoss, and I did
23 the accident sequence precursors analysis, which is
24 designed to address just that question. And,
25 actually, our risk was dominated by the large break

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1 LOCA, coupled with the likely sump failure. And that
2 -- admittedly, the probability of that large LOCA,
3 which is driven by the high end of the corrosion rate
4 curve and then the large blowout, has got a tremendous
5 uncertainty on it. But that becomes a high-risk
6 sequence.

7 And then, medium LOCAs actually was a
8 slightly higher risk sequence, because it also had the
9 CRD and ejection due to that crack growing and getting
10 you. And small LOCA has got a much lower -- better --
11 two order of magnitude lower conditional core damage
12 probability.

13 MEMBER DENNING: Even though there is a
14 question about the high pressure injection, or has
15 that just come about in recirculation and too far out?

16 MR. DeMOSS: Recirculation.

17 MEMBER BONACA: I thought there was an
18 issue with high pressure injection also, Gary, if I
19 recall correctly. There is definitely in
20 recirculation a question on that, and maybe the
21 pressure doesn't hang up long enough.

22 MR. DeMOSS: If I could clarify that. The
23 only issue we had is -- is recirculation, because the
24 pump couldn't pump dirty water and would almost fail
25 with certainty in that situation. And that actually

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1 raised the risk of a small LOCA considerably. It
2 doesn't affect a large LOCA at all, because we don't
3 use that pump in a large LOCA.

4 MR. KIRK: Okay. The next slide,
5 viewgraph 31, compares the forward- and backward-
6 looking analysis in terms of the predicted total LOCA
7 probability on 2-16-02. So, again, the critical
8 difference between the forward- and backward-looking
9 analysis were the backward-looking analysis -- in
10 fact, the inputs that we've provided to Gary for the
11 ASP -- for his ASP work.

12 In the forward-looking analysis, we start
13 with the known condition on 2-16-02 and proceed from
14 there. With the backward-looking analysis, we're
15 required by the ASP protocols to project backward a
16 year's time and make some judgment about what the
17 conditions of the cavity were.

18 And for reasons that we have discussed,
19 there is considerable uncertainty in that. So what we
20 get out of our analysis is that the backward-looking
21 analysis predicts an approximately one in five or 20
22 percent total LOCA probability on 2-16-02 when, in
23 fact, as we know nothing happened.

24 So why are we predicting 20 percent
25 probability? Well, that's, of course, a direct

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1 consequence of the uncertainty regarding the initial
2 conditions that's inherent to that backward-looking
3 calculation.

4 MEMBER POWERS: What you're saying there
5 is that if we had 100 Davis-Besse's of this
6 configuration, 20 of them would have failed a --

7 MR. KIRK: Correct. Yes, that's one
8 possible interpretation.

9 MEMBER POWERS: Well, if the assumptions
10 that went into that calculation are --

11 MR. HISER: Yes, assuming those
12 assumptions represent --

13 MEMBER POWERS: I understand that.

14 MR. HISER: -- the possible range of 20
15 Davis-Besse's.

16 MEMBER POWERS: What you're talking is --
17 with Davis-Besse, if 20 of them would have actually
18 failed, and presumably failed during the operation,
19 they are not shutdown prior to that, day zero.

20 MEMBER DENNING: I'm struggling with
21 exactly what -- when you go -- with a backwards
22 analysis, I can certainly see where you can ask a year
23 earlier, what would the probability -- but how does
24 that, then, impact backwards to today -- I mean, if
25 now today is February of 2002, I mean, we know our

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1 state there. How does the backwards analysis impact
2 that? I mean, you went backwards, and then you did an
3 uncertainty analysis from there coming forwards?
4 It's --

5 MR. KIRK: That's correct.

6 MEMBER DENNING: There's something a
7 little bit --

8 MEMBER POWERS: No, it's the other way
9 around. He knows his state. He knows his actual
10 state today, so we're basically doing it as a Bayesian
11 update. There was probability distribution, right?

12 MR. KIRK: No, I don't think so.

13 MEMBER POWERS: That's not the way you did
14 it, but that's --

15 MR. KIRK: No.

16 MEMBER POWERS: That's what you should
17 have done.

18 MEMBER KRESS: Wanted to know the failure
19 probability as a function of time. That's what the --

20 MEMBER SIEBER: Yes, and we integrated the
21 risk.

22 MEMBER KRESS: I guess the way to get it
23 as a function of time --

24 MEMBER DENNING: In a forward analysis, I
25 mean, that makes a lot of sense -- the forward

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1 analysis. But somehow the backwards analysis --

2 MEMBER KRESS: There is no probability
3 fair of actually the time unaccounted. I want to know
4 how much it was at risk during the time they didn't
5 know about it.

6 MEMBER SIEBER: And the risk keeps
7 changing.

8 MEMBER DENNING: But we know what the
9 state of it is on February 2002, right?

10 MEMBER KRESS: Well, you could say it was
11 always that, but it wasn't.

12 MEMBER DENNING: No, no, I agree. And
13 earlier it was different. But --

14 MR. KIRK: And I think maybe the -- and I
15 have a lot of sympathy for the question you're asking,
16 because it's difficult for me to think about
17 historical events in a probabilistic sense. To me,
18 history is deterministic. but --

19 MEMBER POWERS: I think if you look at in
20 the ensemble --

21 MR. KIRK: Well, it gets to your point.
22 The manifestation that was Davis-Besse did not fail.
23 We know that to be true. But I think perhaps the --
24 whether it's satisfying or not is a different issue.

25 Dr. Powers' representation that, you know,

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1 assuming our assumptions of the conditions a year
2 before date of discovery are reasonable or correct,
3 our calculations are showing that, you know, had there
4 been 100 Davis-Besse -- 100 different evolutions of
5 reality --

6 VICE CHAIRMAN SHACK: Well, I think --

7 MR. KIRK: -- along those lines, roughly
8 one in five of them would have failed.

9 VICE CHAIRMAN SHACK: Your uncertainty in
10 crack growth rate is not as though there is a crack
11 growth rate and you just don't know the answer. You
12 know, there is an aleatory uncertainty in the
13 conditions that could have led to crack growth rates
14 anywhere in there, and so it is an ensemble question.
15 And, you know, it's --

16 MEMBER SIEBER: And to know the
17 probability per reactor year you have to integrate the
18 risk over some period of time to predict it.

19 MEMBER KRESS: That's the reason they want
20 the time.

21 MEMBER SIEBER: Right. That's why you go
22 back.

23 MR. KIRK: Yes. Just, you know, taking
24 those results apart a little bit more into the
25 different LOCA types, I'll just point out that even

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1 though the graphs that I'm showing you, and the graphs
2 that appear in our reports, go both for the backward-
3 looking analysis a year before day of discovery and
4 then out to we're predicting a total LOCA probability
5 of unity.

6 The ASP analysis only actually used the
7 predicted LOCA probabilities in the year before the
8 day of discovery. So all those other LOCA
9 probabilities are just shown for information purposes
10 only. It's not something that ever actually got used
11 in the analysis.

12 And, again, you know, focusing attention
13 on the year before day of discovery, as was the case
14 with the forward-looking analysis, a small break LOCA
15 is, again, the most likely outcome, although as Gary
16 has pointed out from an integrated risk perspective,
17 that is not what is dominating the risk.

18 And just to look at the effects of the
19 different modeling assumptions that we made, which
20 basically includes how we selected statistical
21 distributions to represent the key variables in our
22 analyses, on the day of discovery our backward-looking
23 analysis is predicting a best estimate total LOCA
24 probability of about 20 percent. And that has a
25 range, depending upon how we statistically represented

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1 our modeling assumptions, of between 14 and 24
2 percent.

3 If you look at small break LOCA
4 probability, best estimate is 18 percent ranging from
5 2 to 18; medium break, 1 to 15 percent with the best
6 estimate being 1; and large break anywhere from 0 to
7 9 percent with the best estimate of about 3.

8 And, again, I know these are some -- some
9 fairly substantial ranges, but given the uncertainties
10 involved and the limited state of knowledge that's
11 what you wind up with.

12 So to summarize -- go ahead. I'm sorry.

13 MEMBER POWERS: You have avoided putting
14 your probability access on a logarithm stage, so we
15 can see at what point you started crossing our level
16 of pain with respect to vessel integrity.

17 MR. KIRK: Well, I think the -- maybe the
18 answer is -- you're looking for is more fairly dealt
19 with in Gary's analysis, and I would just point out
20 that the curves I'm showing you here are merely --
21 they're the output of our structural calculation and
22 form inputs to Gary's analysis, where those type of
23 issues are taken up in a much more sound, scientific
24 way.

25 MR. HISER: Yes. I think maybe one

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1 message from this is the numbers are huge. You know,
2 they're orders of magnitude higher than what's
3 acceptable. So you need to do proper maintenance and
4 not allow these kinds of conditions to occur. I mean,
5 a lot gets --

6 MEMBER POWERS: Is it a good idea to have
7 a hole in the head like that?

8 MEMBER DENNING: Well, the thing that
9 strikes me is the forward analysis -- to me, the
10 forward analysis says that if we had buttoned it up
11 and operated for the next cycle that it probably would
12 have had a break.

13 MR. HISER: Probably. Yes, probably.
14 They were on a two-year cycle and, what, the 95
15 percentile was 22 months. So that's pretty close to
16 one.

17 MR. KIRK: Unless you take Member Kress'
18 very optimistic view that the corrosion has stopped.

19 MEMBER DENNING: But he didn't make that
20 view. He just said that -- he just said that it
21 necessarily --

22 MEMBER POWERS: I thought you believed it
23 passionately.

24 MEMBER KRESS: Well, I do hold the view.

25 VICE CHAIRMAN SHACK: Yes. But what's

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1 your probability of belief in that number?

2 MEMBER KRESS: .8.

3 VICE CHAIRMAN SHACK: Oooh. He's
4 convinced.

5 MR. KIRK: Okay. So just to -- to
6 summarize for our analysis of the as-found condition,
7 our forensic examinations, and those performed by
8 others, most notably BWXT found no ductile tearing
9 initiated from the corrosion-assisted flaws, and that
10 suggests that cladding rupture was in no way imminent
11 on the day of discovery.

12 Our analysis predicted that there was no
13 crack initiation on the day of discovery, and our
14 analysis also quantified that pressure in excess of
15 the relief value setpoint would have been needed to
16 rupture the cladding on 2-16-02.

17 Our forward-looking analysis where we
18 treat that as-found condition is known, and try to
19 maintain some insight into events in the future --
20 said that we had between 2 and 22 months more of
21 operation that would have been needed at the operating
22 pressure to rupture the cladding. And the best
23 estimate, meaning the median value, is somewhere
24 around five months.

25 And the most likely consequence of

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1 cladding rupture would have been a small break LOCA.
2 Using the backward-looking analysis that was input
3 into the ASP calculation, and taking a single point
4 away from that, we're predicting approximately a one
5 in five chance of some sort of LOCA on the day of
6 discovery, and in all likelihood that would have been
7 a small break.

8 MEMBER KRESS: Your conclusion that the
9 most likely consequence is a small break LOCA implies
10 to me that the failure was crack growth.

11 MR. KIRK: Yes. What was? I'm sorry.

12 MEMBER KRESS: That the failure mechanism
13 was actually crack growth.

14 MR. KIRK: Yes, yes.

15 MEMBER KRESS: So that my -- my position
16 that the cavity didn't change in size much doesn't
17 really affect that the vessel probably would have
18 failed anyway, because of crack growth and --

19 MR. KIRK: Yes. It's --

20 MEMBER KRESS: -- it probably wouldn't
21 have been much different in timing.

22 MR. KIRK: Well, obviously, it's a race.
23 As the cavity size gets bigger, you get a bigger
24 unbacked area, so you get more bending stress.

25 MEMBER KRESS: So the crack grows faster

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1 is the --

2 MR. KIRK: So there is more -- yes, there
3 is more applied stress, but there is already an
4 unbacked area, so there is already bending.

5 MEMBER KRESS: Yes.

6 MR. KIRK: So the cracks are, I think safe
7 to say, already growing.

8 MEMBER KRESS: Yes.

9 MR. KIRK: So, yes, you're right. Even if
10 the cavity had stopped growing entirely, that doesn't
11 mean that it wouldn't have failed, at least in my
12 view.

13 VICE CHAIRMAN SHACK: No. I mean, you
14 could actually do that calculation, presumably.

15 MR. KIRK: Yes. Well, presumably, it's
16 one of the many thousands of manifestations that we
17 did.

18 MEMBER KRESS: Yes, it would have --

19 MR. KIRK: But, I mean, you'd shift the
20 probabilities if you turned it off.

21 MEMBER KRESS: But I'll bet the time
22 doesn't change that much.

23 MR. KIRK: Probably not.

24 MEMBER KRESS: For failure.

25 MR. KIRK: That's it.

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1 MEMBER SIEBER: Do any of the members have
2 additional questions or comments?

3 MEMBER KRESS: Comment. That was a
4 terrific presentation. I appreciate it.

5 MEMBER POWERS: I reiterate that had you
6 been -- had this work been submitted for the quality
7 review, I think it would have scored extraordinarily
8 highly.

9 MEMBER SIEBER: Yes.

10 MEMBER KRESS: So we appreciate that.
11 Thank you.

12 MR. SCOTT: Jack?

13 MEMBER SIEBER: Yes.

14 MR. SCOTT: I'd like to ask a question if
15 I could. Can you speak a little bit to the
16 probability of a rod ejection having occurred in
17 conjunction with this? Could you all address that?

18 MR. KIRK: No, I personally can't. Can
19 anybody else? Sorry, just not my area.

20 MR. SCOTT: Okay.

21 MR. DeMOSS: Yes. Gary DeMoss, the ASP
22 panelist again. The rod ejection probability was
23 considered in the ASP analysis -- separate materials
24 -- an analysis which I'm not equipped to speak in
25 detail on. I'm a PRA guy.

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1 But it was actually the most likely cause
2 of a medium LOCA, more likely than the ejection of the
3 cladding that they -- analogous to what they showed
4 from the laboratory.

5 MR. SCOTT: Gary, do you know what the
6 probability they came up with was on a rod ejection
7 occurring?

8 MR. DeMOSS: I could dig it out here I
9 think fairly quickly. But it was -- it was higher
10 than the 1 percent medium LOCA that was generated for
11 the unbacked cladding.

12 MR. SCOTT: So the medium break LOCA
13 that's in the presentation here does not include the
14 rod ejection situation.

15 MR. DeMOSS: No. No, that's just cladding
16 -- cladding failure.

17 MR. SCOTT: Okay.

18 MR. DeMOSS: Two percent was the estimate
19 of the -- with an analogous -- analogously constructed
20 analysis that -- you have a 2 percent change of rod
21 ejection in that -- during that year leading up to the
22 discovery of the problem.

23 MR. SCOTT: Okay. thank you.

24 VICE CHAIRMAN SHACK: Wouldn't the rod
25 ejection require the whole failure of the nozzle?

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1 MEMBER SIEBER: Yes.

2 VICE CHAIRMAN SHACK: I mean, that would
3 seem a whole lot less likely.

4 MEMBER SIEBER: I would think so.
5 Certainly not the --

6 MR. HISER: Given that it wasn't a large
7 circ crack.

8 MEMBER SIEBER: Yes.

9 VICE CHAIRMAN SHACK: I guess until you
10 know how they got to that number, I guess, you know,
11 I mean, if you had postulated the possibility of a
12 circ crack forming --

13 MEMBER SIEBER: That's another analysis,
14 however, which I don't think has been done. Right?

15 MR. KIRK: No. I don't believe so.

16 MEMBER RANSOM: What does the ASP stand
17 for?

18 MR. KIRK: Accident sequence precursor.

19 MEMBER RANSOM: What is it?

20 MR. KIRK: Accident sequence precursor.

21 MEMBER RANSOM: Oh, okay.

22 MR. DeMOSS: Let me make a correction.
23 I've reread my analysis. One percent, not 2 percent,
24 is the chance of a rod ejection. Just still higher
25 than maybe you accept, but it's based on the work done

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1 for the SDP.

2 VICE CHAIRMAN SHACK: But do you know if
3 that is based on some sort of an estimate of a
4 circular crack?

5 MR. DeMOSS: Yes, that's based on Steve
6 Long's work to estimate the circular crack.

7 VICE CHAIRMAN SHACK: And I know what he's
8 relying on.

9 (Laughter.)

10 We know how shaky that analysis is.

11 MEMBER SIEBER: Well, which way is it
12 shaking?

13 MR. DeMOSS: Shaking would not be a good
14 thing.

15 MEMBER SIEBER: Are there any additional
16 questions? If not, I would like -- I thought it was
17 a good presentation and a good analysis by the heavy
18 section steel folks at Oak Ridge. And thanks, Mark
19 and Alan, for putting this together for us.

20 I understand you're not expecting a
21 written response from us, unless we feel it necessary
22 for -- to do so.

23 MR. HISER: In all honesty, we're hoping
24 this is the last time we have to talk about Davis-
25 Besse, cladding, and calculations.

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1 MEMBER SIEBER: I probably hope that more
2 than you do.

3 MR. HISER: Oh no. Not a chance.

4 (Laughter.)

5 Not a chance.

6 MEMBER SIEBER: Well, I'm hoping that
7 there are no problems like that to analyze in the
8 future. Okay?

9 So thank you very much. And, Mr.
10 Chairman, I turn it back to you.

11 VICE CHAIRMAN SHACK: Okay. We're on time
12 again. And, actually, we can come back early since
13 we're on our own at this point. So --

14 MEMBER DENNING: George would like to say
15 he'd like a full 15 minutes.

16 (Laughter.)

17 MEMBER APOSTOLAKIS: Yes, I would like a
18 full 15 minutes.

19 VICE CHAIRMAN SHACK: Then we'll go to
20 4:15. We'll live it up. Well, be back at 4:10.

21 (Whereupon, at 3:51 p.m., the proceedings
22 in the foregoing matter went off the
23 record.)

24

25

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on
Reactor Safeguards
526TH Meeting

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.



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An Assessment of the Structural Integrity Challenge Posed by Boric Acid Wastage in the Davis Besse RPV Head



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**B. Richard Bass, Paul Williams,
Wally McAfee, and Sean Yin**

Oak Ridge National Laboratory

ACRS Briefing

USNRC Headquarters • Rockville, MD • 6th October 2005

VG 1

Objectives of Our Analyses

- **As Found (only possible reality benchmark)**
 - Assess the structural integrity of the primary reactor coolant pressure boundary for the conditions that existed at Davis Besse on February 16, 2002
- **Looking forward (SDP support)**
 - Assess the structural integrity of the primary reactor coolant pressure boundary for conditions postulated to exist at Davis Besse had it not been taken off-line for a scheduled maintenance outage on February 16, 2002
- **Looking backward (ASP support)**
 - Assess the structural integrity of the primary reactor coolant pressure boundary for conditions postulated to exist at Davis Besse for February 16, 2002 minus one year (ASP analysis)

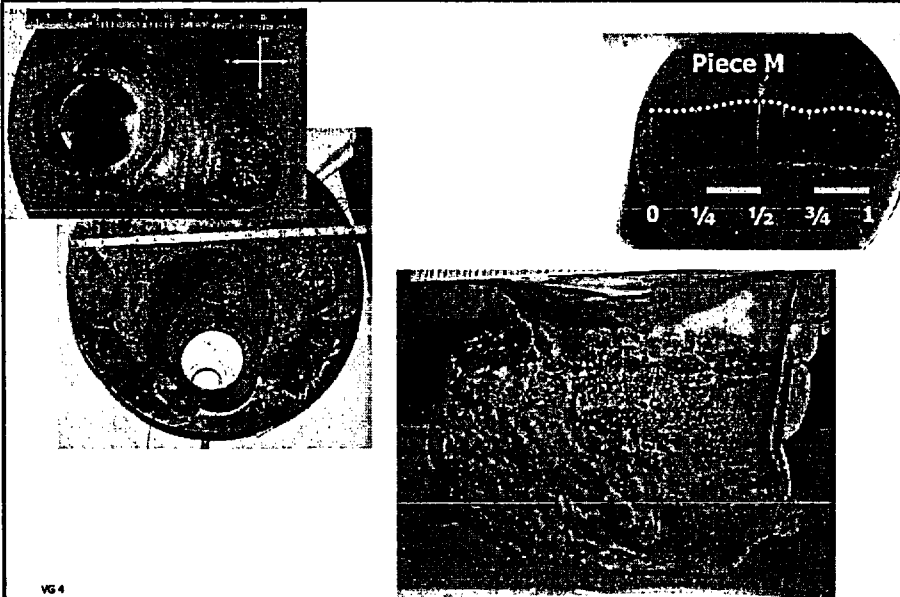
VG 2

Presentation Outline

- **Description of the as found state**
- **As Found analysis**
 - **Methodology**
 - **Results**
- **Forward & backward looking analyses**
 - **Methodology**
 - **Results**

VG 3

16th Feb 02 Conditions at Davis Besse (*"as found"*)

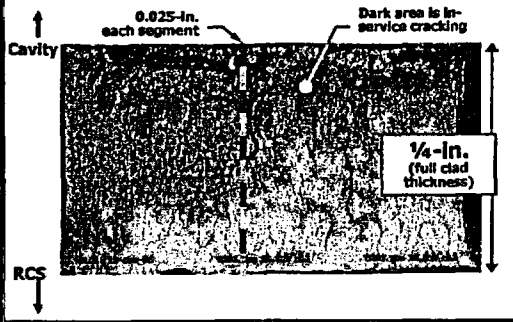


VG 4

Davis Besse Crack Characterization

Crack Depth

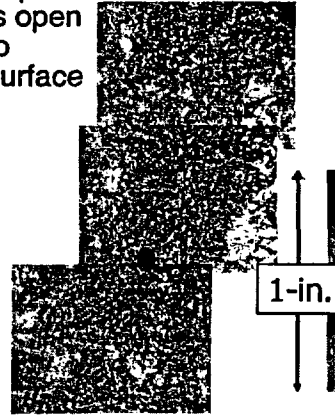
- 0.1-in max (over 20% of length)
- 0.065-in. average



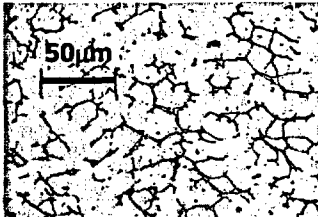
vg 5 [All metallography by J. Hyres, BWXT]

Crack Length

- 2-in max
- Central 0.66-in. has significant depth & is open to surface



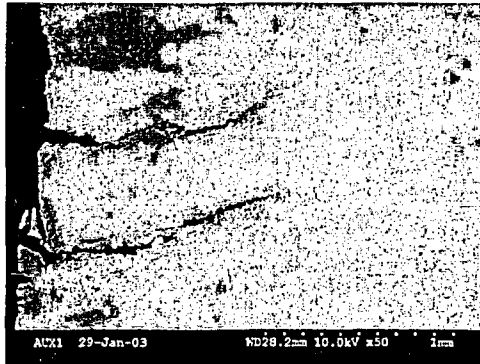
Crack Extension Mechanism



Dendritic solidification structure in 308SS

- Dark = ferrite
- Light = austenite

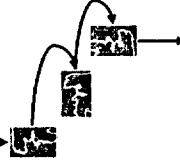
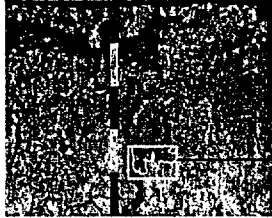
Cracks in the 308 stainless steel cladding formed when the concentrated boric acid solution in the cavity preferentially dissolved the ferrite phase. Thus, the cracking is intergranular (i.e., between the austenite grains).



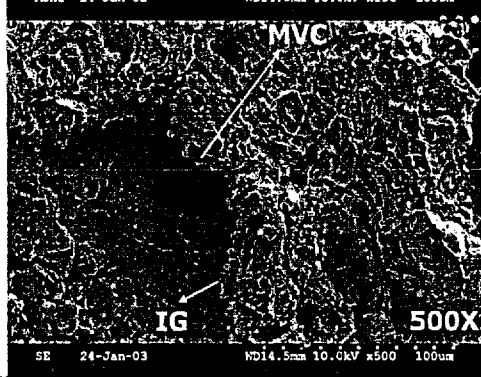
AUX1 29-Jan-03 WD28.2mm 10.0kV x50 1mm

vg 6

Crack Morphology

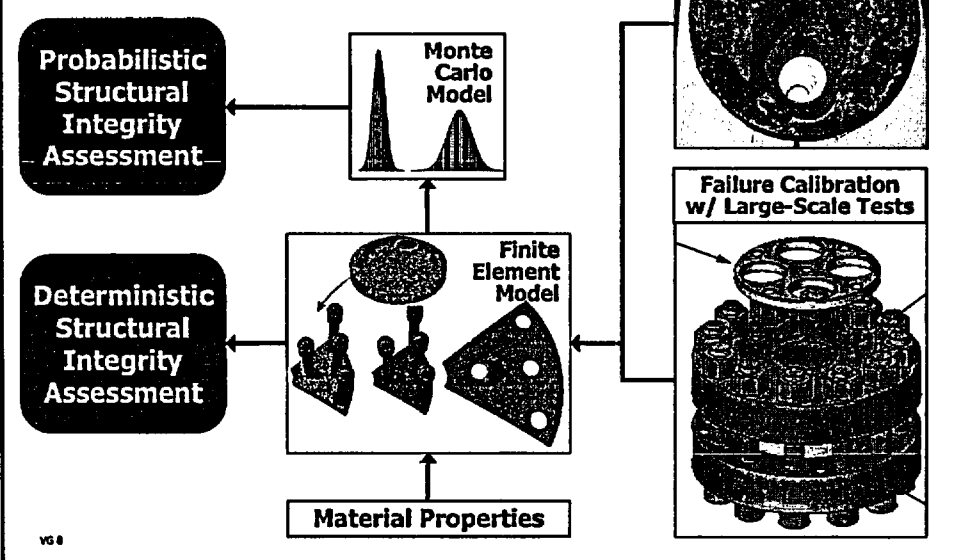


- In-service cracks are intergranular & result from preferential attack of ferrite phase by boric acid.
- Lack of ductile tearing on service darkened side of fracture suggests that operating pressure did not load cladding above the ductile crack initiation threshold (i.e., cladding rupture was not imminent on 16 Feb 02).



VG 7

Methodology for Integrity Assessment of the "As Found" State



VG 8

Input Information

- **As-found configuration**

- Cavity geometry
- Crack size and morphology

While described on the preceding slides, the totality of this information was not available at the outset

- ✓ June 2003: BWXT failure analysis report (for Framatome ANP)
- ✓ April 2004: BWXT report on detailed crack examination (for ORNL)

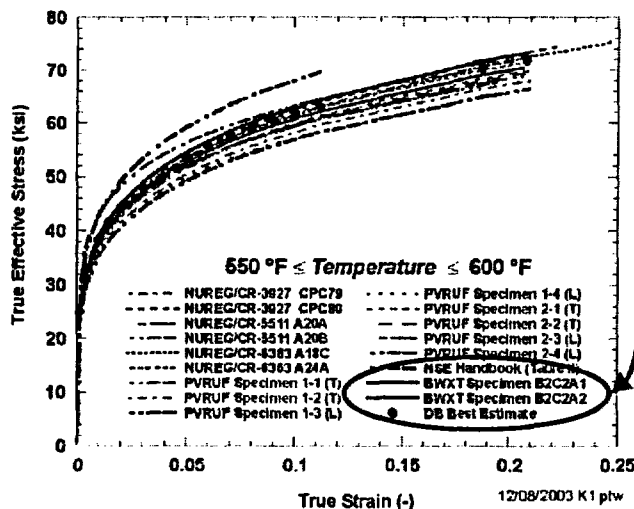
- **Cladding strength & fracture toughness properties**

- **Cladding failure mode & predictive benchmark**

- Discerned based on burst testing

VG 9

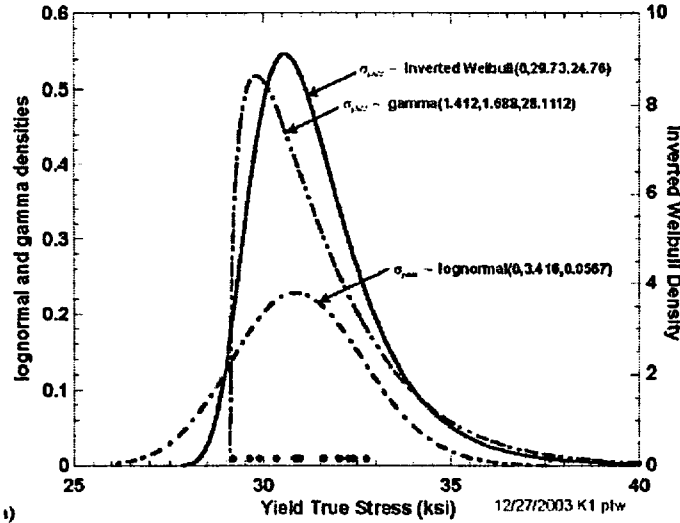
Cladding Strength



Tests on DB cladding agree well with literature data

VG 10

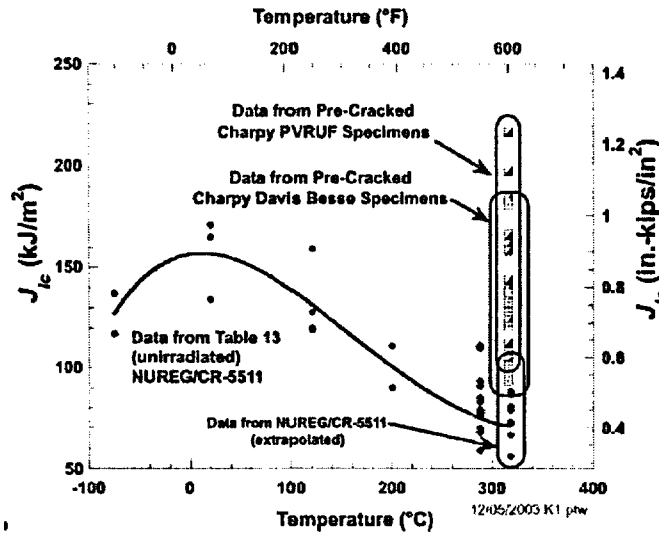
Cladding Strength



These data provide the basis for a statistical description used in our Monte Carlo analysis.

i)
VG 11

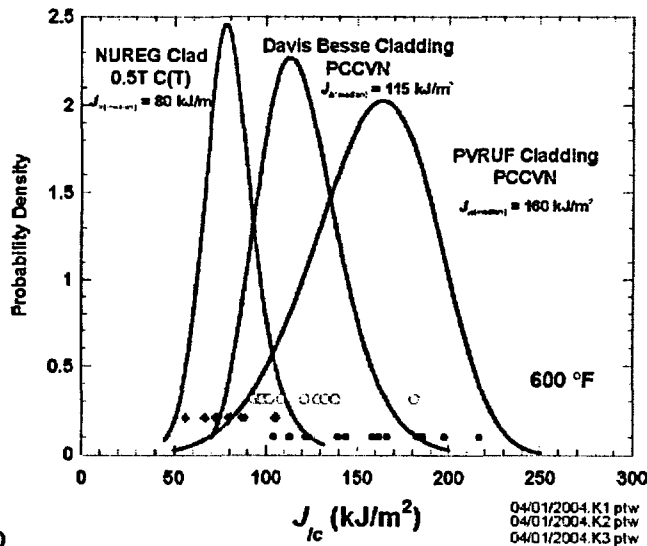
Cladding Fracture Toughness



Tests on DB cladding agree with literature data

i)
VG 12

Cladding Fracture Toughness



These data provide the basis for a statistical description used in our Monte Carlo analysis.

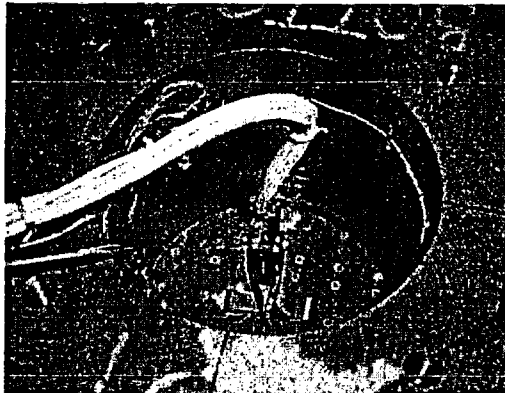
Burst Testing

- Captures the essential structural characteristics of state of DB on 2-16-02
 - Un-backed cladding area
 - Flaw depth

▪ Not intended as a 1:1 model or representation of DB

- Objectives of tests
 - Validate opinion that cladding will fail by ductile tearing
 - Asses accuracy / conservatism in model predictions of failure loads

Pre-Test



VG 14

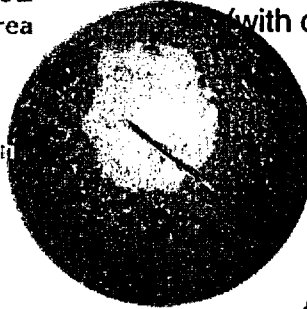
Burst Testing

- Captures the essential structural characteristics of state of DB on 2-16-02
 - Un-backed cladding area
 - Flaw depth

- Not intended as a 1:1 model or representation of DB

- Objectives of tests
 - Validate opinion that cladding will fail by ductile tearing
 - Asses accuracy / conservatism in model predictions of failure loads

POST Test (with crack)



VG 15

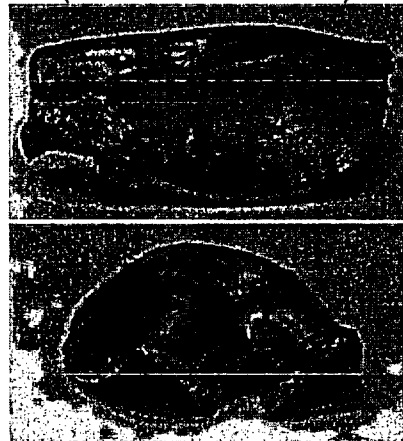
Burst Testing

- Captures the essential structural characteristics of state of DB on 2-16-02
 - Un-backed cladding area
 - Flaw depth

- Not intended as a 1:1 model or representation of DB

- Objectives of tests
 - Validate opinion that cladding will fail by ductile tearing
 - Asses accuracy / conservatism in model predictions of failure loads

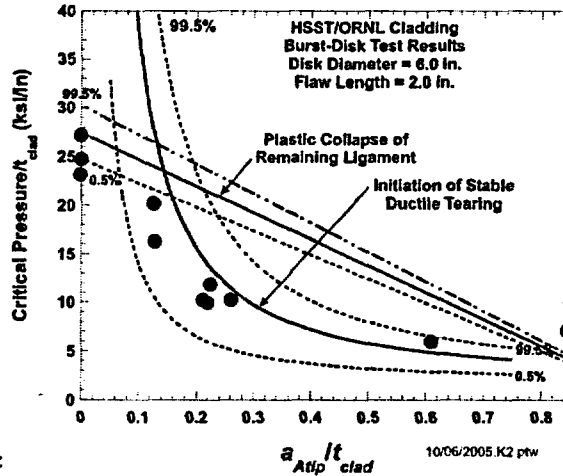
POST Test (no or shallow crack)



VG 16

Comparison of Burst Test Data with Model Predictions

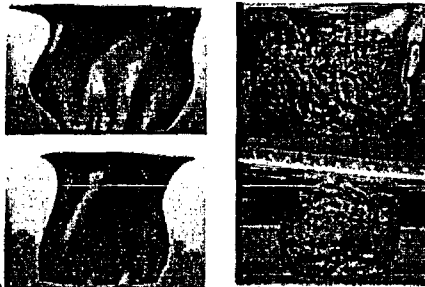
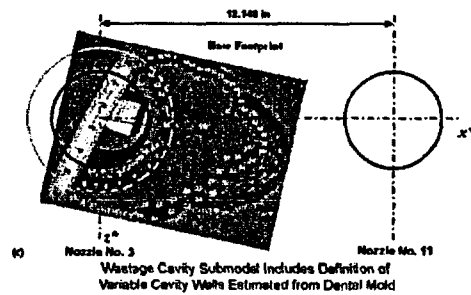
- Good test-to-test repeatability
- For deeper cracks (> ≈15% of the cladding thickness)
 - Stable failures (leaks)
 - Predicted well by ductile cracking model
- For shallower cracks or no cracks
 - Catastrophic failures (blowout of un-backed area)
 - Predicted well by plastic collapse model



- ✓ Failure mode characterized
- ✓ Model validated

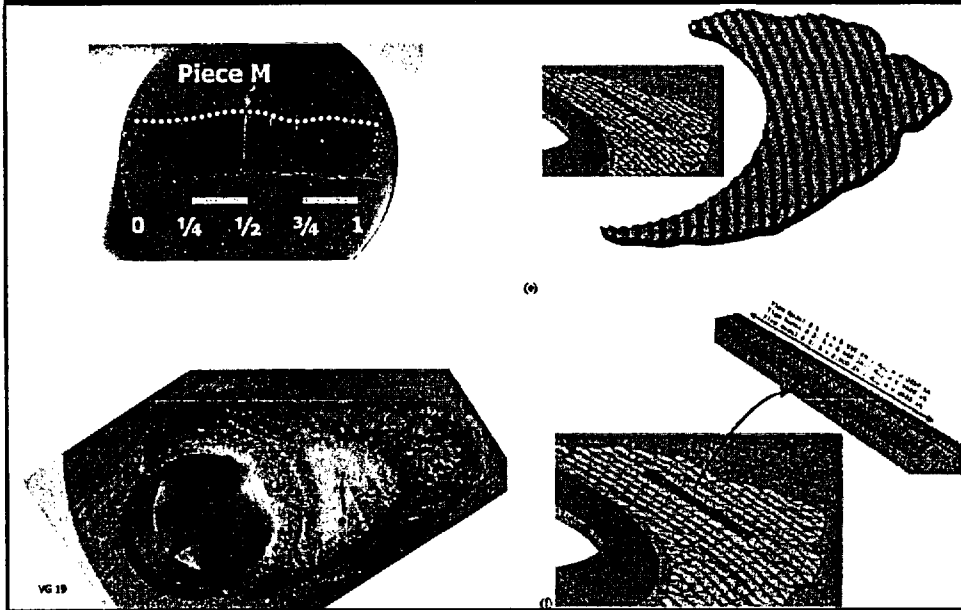
VG 17

Geometric Inputs to Finite Element Model of As-Found State

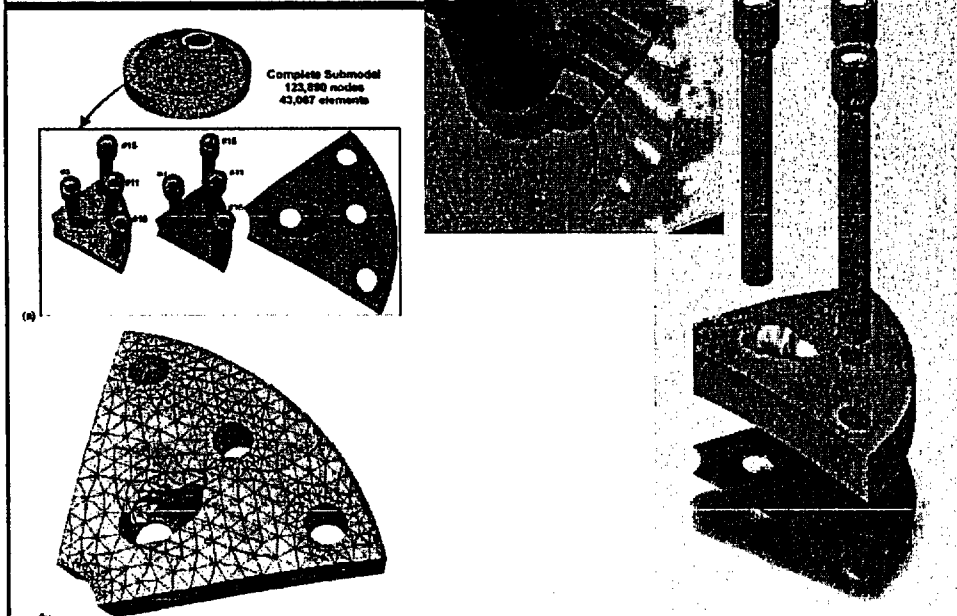


VG 18

Geometric Inputs to Finite Element Model of the As-Found State



Finite Element Model of As-Found State



As-Found Analysis

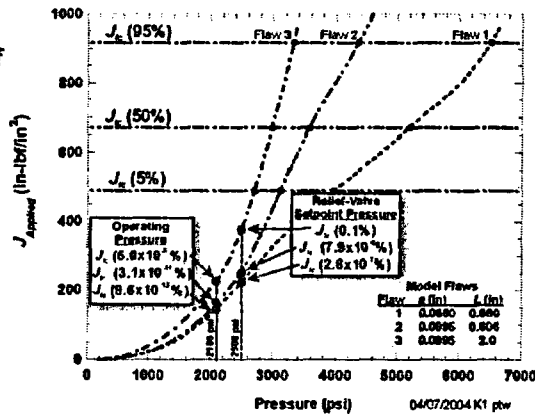
- is based on the following
 - Geometrically accurate finite element model to estimate stresses
 - Actual DB properties for cladding and ferritic steel strength
 - Actual DB properties for cladding toughness
 - 3 Different idealizations of as-found crack network
 - ✓ Flaw 1: Best-estimate depth (0.065-in.) & length (0.66-in.)
 - ✓ Flaw 2: Bounding depth (0.1-in.) & best estimate length (0.66-in.)
 - ✓ Flaw 3: Bounding depth (0.1-in.) & length (2-in.)

VG 21

As-Found Analysis Results

Prediction

- Pressure in excess of relief valve setpoint pressure needed to rupture cladding
- Factor of at least 1¼ safety margin exists against ductile crack initiation occurring at the operating pressure even assuming
 - ✓ A bounding flaw characterization
 - ✓ A lower bound fracture toughness characterization



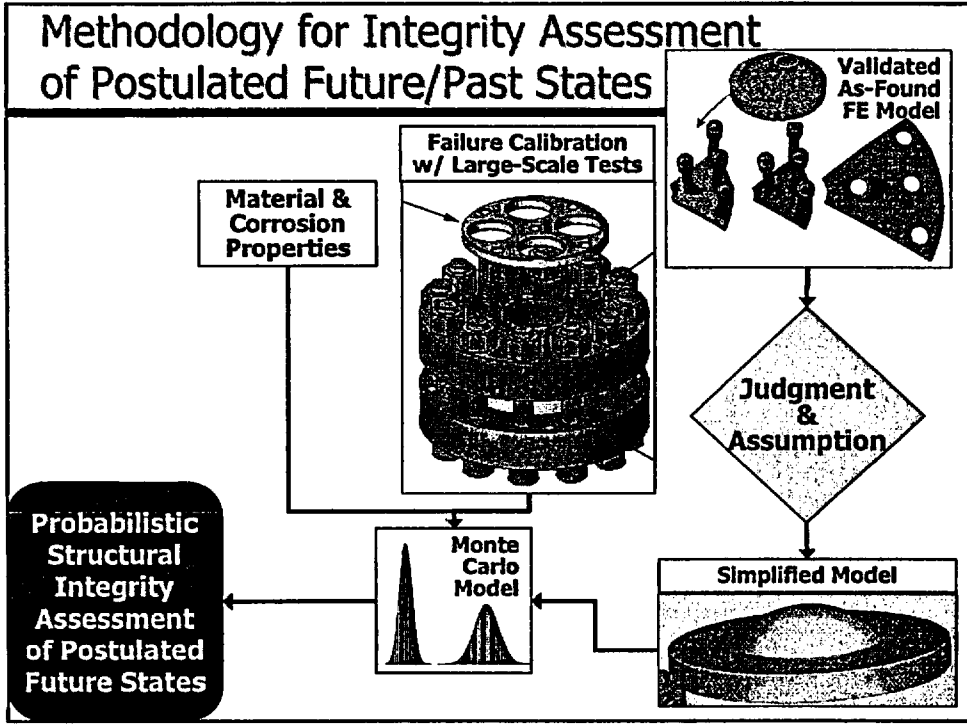
Our prediction is that

- Failure did not occur
- Ductile crack initiation from crack network did not occur

What actually happened was

- Failure did not occur
- Ductile crack initiation from crack network did not occur

VG 22



Assumption

- *The complex cavity shape can be modeled as a circle*
 - For failure by plastic collapse, un-backed cladding area is much more important than cavity shape
 - For failure by ductile tearing, circular assumption is conservative
 - State of knowledge was (and is) not adequate to defend a more complex representation of how the cavity shape evolved (or would have evolved) with time

Chakrabarty and Alexander (1970) Theory Compared to FEM Predictions

Wastage Cavity Footprint Area (in ²)	Estimated Burst Pressure (ksi)	Notes
50	8.5	"As-Found" Model Feb. 2004
100	6.5	"As-Found" Model Sept. 2003
150	4.5	Circular Diaphragms (Center-Membrane Failure Theory)

Ellipsoidal

Self-similar

02/02/2004 K1 p4v

Input Information

- **Based on statistical representations of data**
 - Toughness properties
 - Strength properties
- **Based on engineering judgments**
 - LOCA binning rules
 - Statistical fitting of data
- **Based on expert opinions benchmarked to data**
 - General corrosion properties of the ferritic RPV steel (controls cavity growth rate)
 - Corrosion crack growth properties of the austenitic cladding (controls crack depth growth in cladding)
- **Based on expert opinion (for ASP analysis only)**
 - Crack depth on 2-16-02 *minus one year*
 - Cavity size on 2-16-02 *minus one year*

Discussed
already

VG 25

Engineering Judgments

- **LOCA binning rules**
 - Small up to 3.5" diameter
 - Medium from 3.5" – 4.8" diameter
 - Large > 4.8" diameter
 - "Conservative" model equates thru-clad cracking with failure of cladding
 - "Best estimate" model assesses stability of thru-clad crack based on standard *J-R* curve analysis techniques
- **Statistical fitting of data**
 - Illustrated on the following slides

VG 26

Expert Elicitation Information

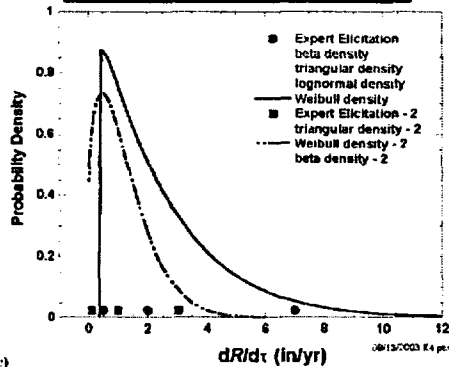
Judgments Obtained from Informal Expert Elicitation

Parameter Description	Parameter Units	Subject Matter Expert ID No.#	Lower Bound	Median (BE) Value	Upper Bound
			Associated Percentiles		
			5%	50%	95%/99.9%
Effective Cavity		#1	0	1.25	2
Radius at TOD-1	(in.)	#2	0	1.125	2.5
R_0		#3	0	1.5	2.25
Effective Cavity		#1	1	2	7
Wastage Rate	(in./year)	#2	0.5	2	6
dR/dt		#3	0.75	1.5	(-)
Flaw Initiation		#1	12	36	48
Time w.r.t. TOD	(months)	#2	1	6	60
$\Delta t_{flaw-init}$		#3	(-)	(-)	(-)
Effective Flaw		#1	0.001	0.01	0.1
Growth Rate	(in./month)	#2	0.001	0.01	0.1
da/dt		#3	0.004	0.01	0.04

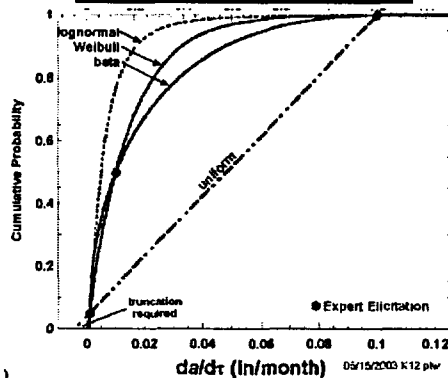
VG 27

Statistical Representation of Expert Elicitation Information (Examples)

Cavity Growth Rate



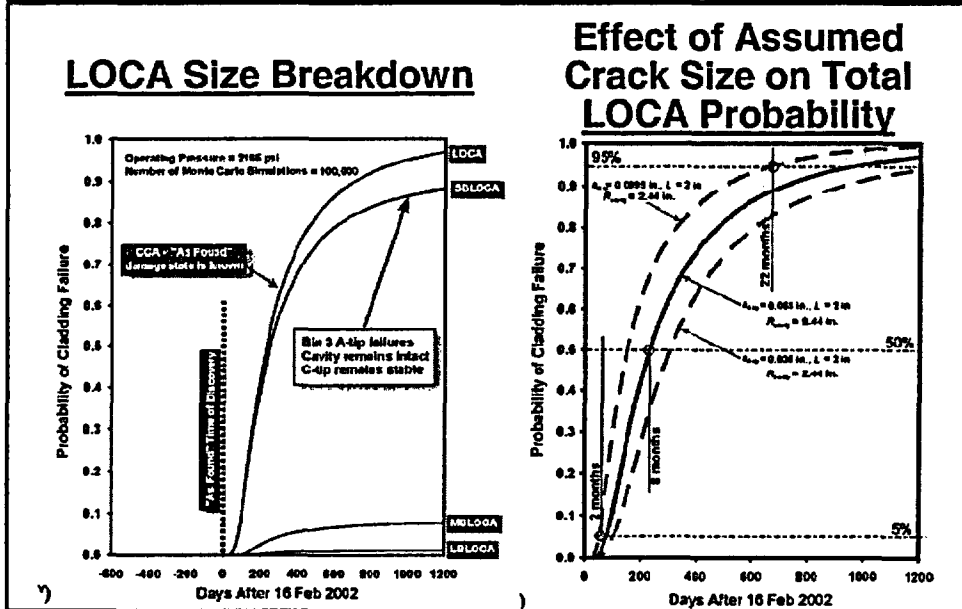
Crack Growth Rate



Sampling Distributions Categories	dR/dt (in/yr)	da/dt (in/month)	R_0 (in)	$\Delta t_{flaw-init}$ (months)
BE	beta-1	Weibull	beta	Weibull
MC	triangular	lognormal	triangular	lognormal
LC	Weibull	uniform	normal	triangular

VG 28

Results of "Forward Looking" Analysis (as-found state known)

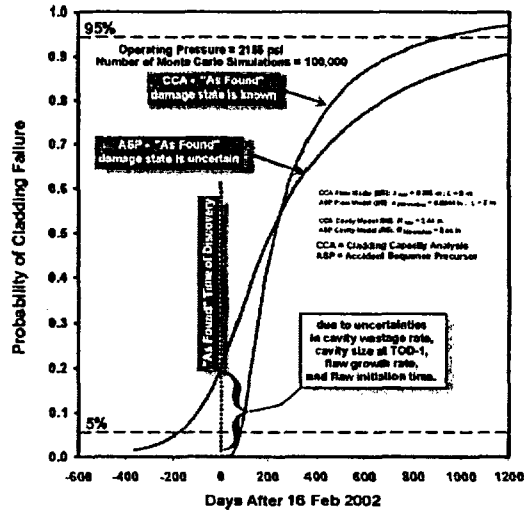


Results of "Forward Looking" Analysis (as-found state known)

- Based on bounding flaw model (Flaw 3: consistent with ASME practice) between 2 & 22 months of operation beyond 2-16-02 could have taken place before the cladding was compromised
 - Best estimate is 5 months
- Had the primary pressure boundary been compromised the most likely consequence was a small break LOCA
 - Known deep initial flaw depth favors SB-LOCA

Comparison of Forward and Backward Looking Analysis on the Predicted Total LOCA Probability on 2-16-02

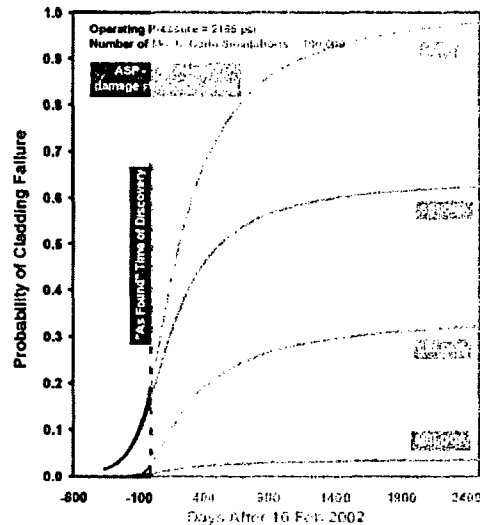
- Backward looking analysis predicts an $\approx 20\%$ LOCA probability on 2-16-02 (when, in fact, no LOCA occurred)
- This occurs as a direct consequence of uncertainty regarding the initial conditions assumed in the backward looking analysis, i.e.
 - Cavity size on 2-16-01
 - Depth of flaws in cladding on 2-16-01



VG 31

Summary of Best-Estimate "Backward Looking" (ASP) Analysis

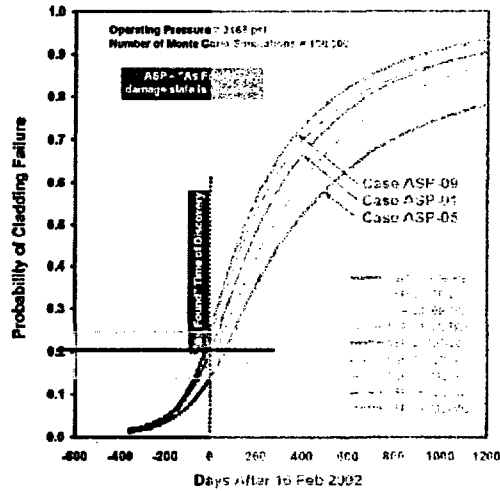
- ASP analysis only uses predicted LOCA probabilities between 2-16-01 and 2-16-02
- LOCA probabilities for dates beyond 2-16-02 shown *for information only*
- As was the case with the forward looking analysis, SB-LOCA is the most likely outcome had the pressure boundary been compromised



VG 32

Effect of Uncertainties / Judgments on Total LOCA Probabilities

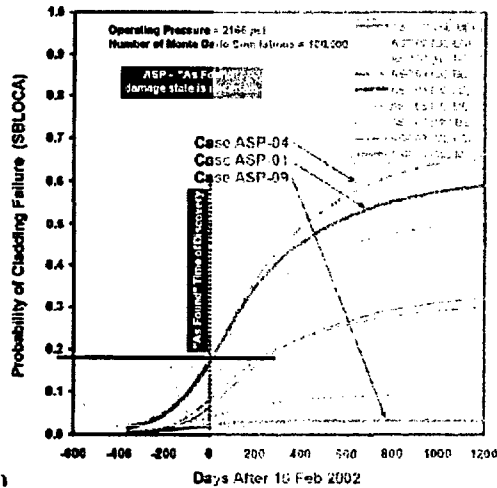
- **Total LOCA probability on 2-16-02**
 - **Min** ≈ 14%
 - **Max** ≈ 24%
 - **Best estimate** ≈ 20%



VG 33

Effect of Uncertainties / Judgments on SB-LOCA Probabilities

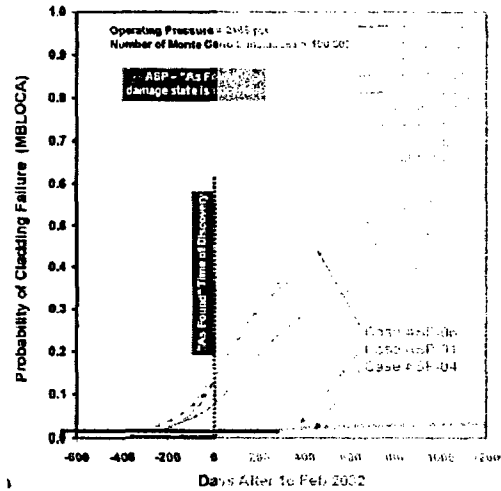
- **SB-LOCA probability on 2-16-02**
 - **Min** ≈ 2%
 - **Max** ≈ 18%
 - **Best estimate** ≈ 18%



VG 34

Effect of Uncertainties / Judgments on MB-LOCA Probabilities

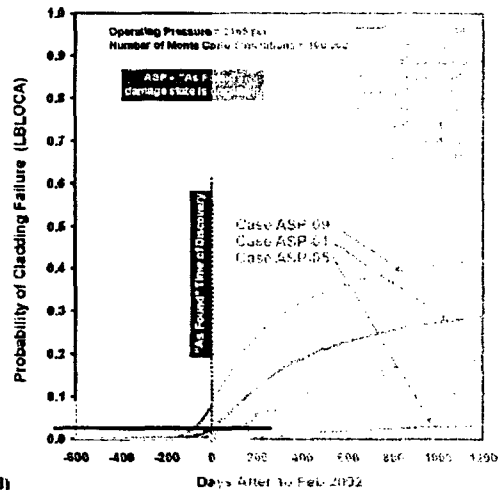
- MB-LOCA probability on 2-16-02
 - Min < 1%
 - Max ≈ 15%
 - *Best estimate* < 1%



VG 35

Effect of Uncertainties / Judgments on LB-LOCA Probabilities

- LB-LOCA probability on 2-16-02
 - Min = 0%
 - Max ≈ 9%
 - *Best estimate* ≈ 3%

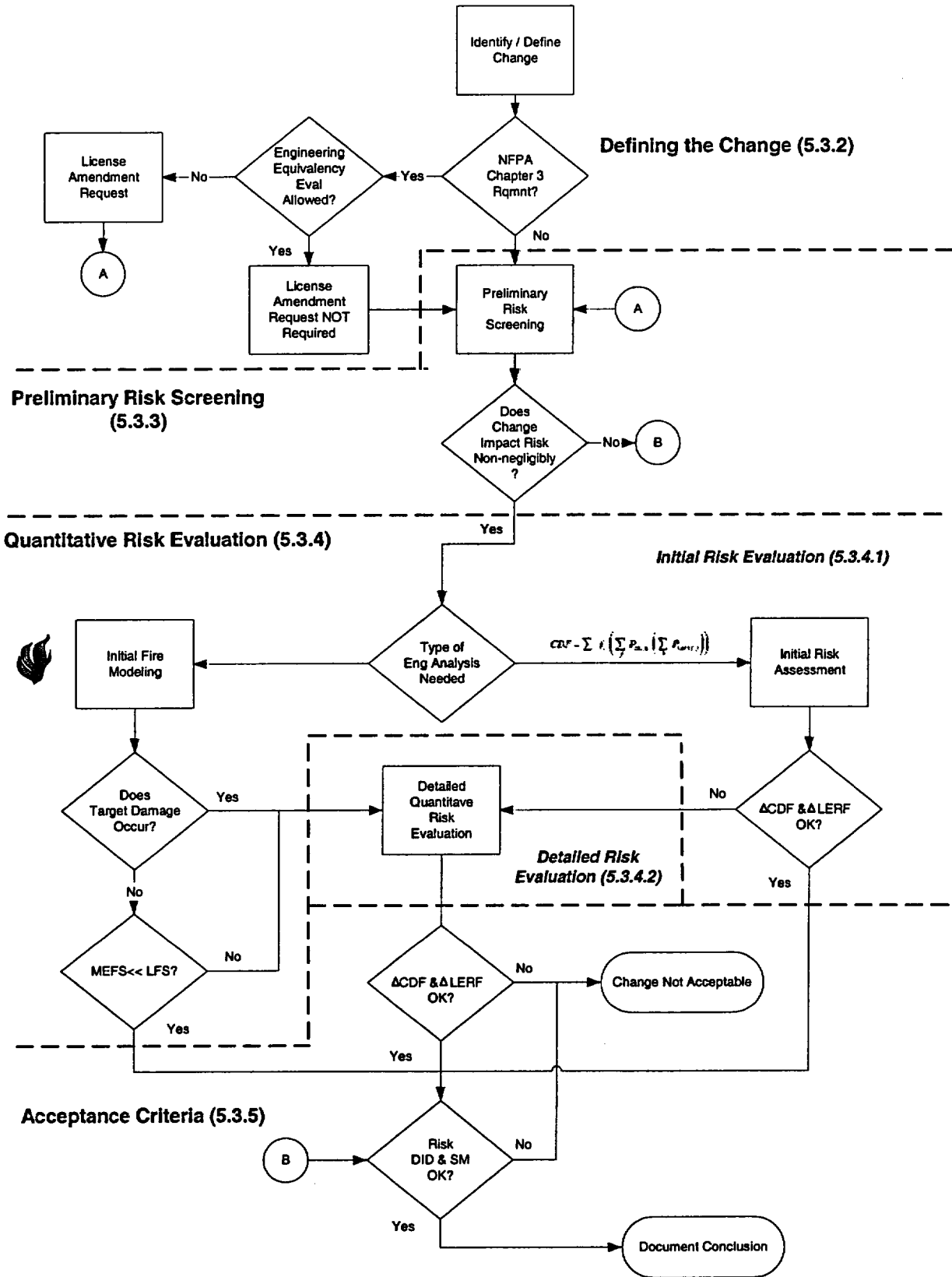


VG 36

Summary

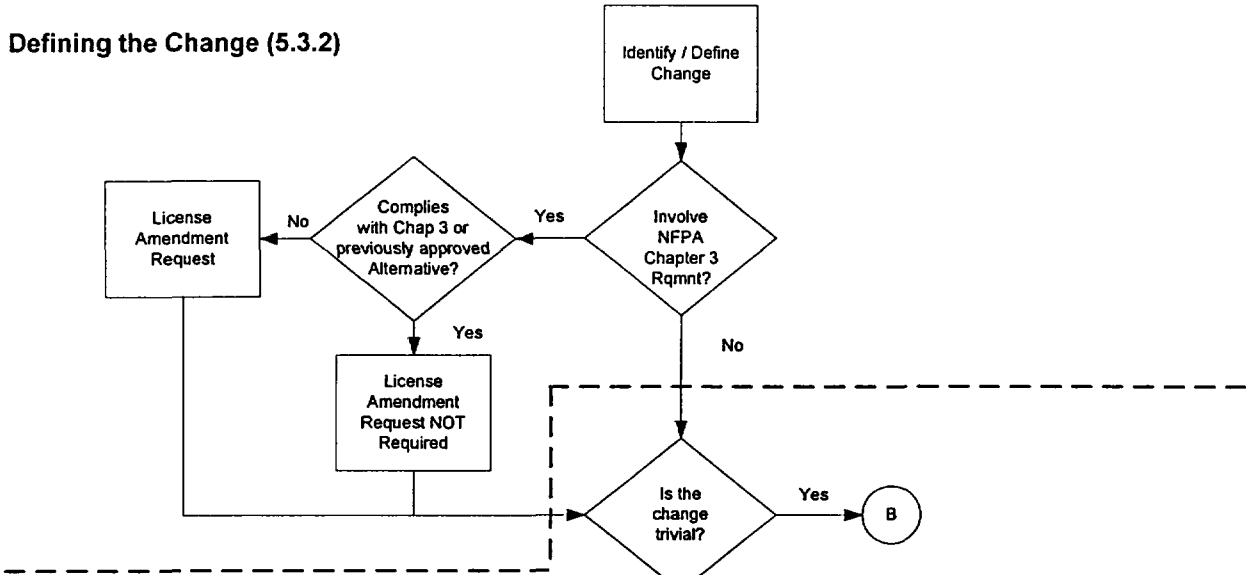
- **As-found analysis**
 - Forensic exams found no ductile tearing initiated from corrosion assisted flaws – suggests cladding rupture was not imminent
 - No crack initiation predicted on day of discovery
 - Pressure in excess of relief valve setpoint pressure would have been needed to rupture the cladding
- **Forward looking analysis (as found condition known)**
 - 2-22 months more operation needed to rupture the cladding ... best estimate is ~5 months.
 - Most likely consequence of cladding rupture is SB-LOCA
- **Backward looking analysis ... ASP (as found condition uncertain)**
 - ~20% (+/-5%) total LOCA probability on day of discovery
 - Most likely consequence of cladding rupture is SB-LOCA

VG 37

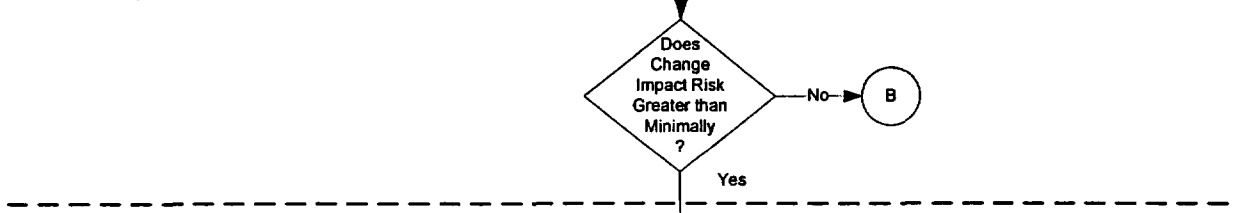


NEI 04-02 Figure 5-1 Revision 0

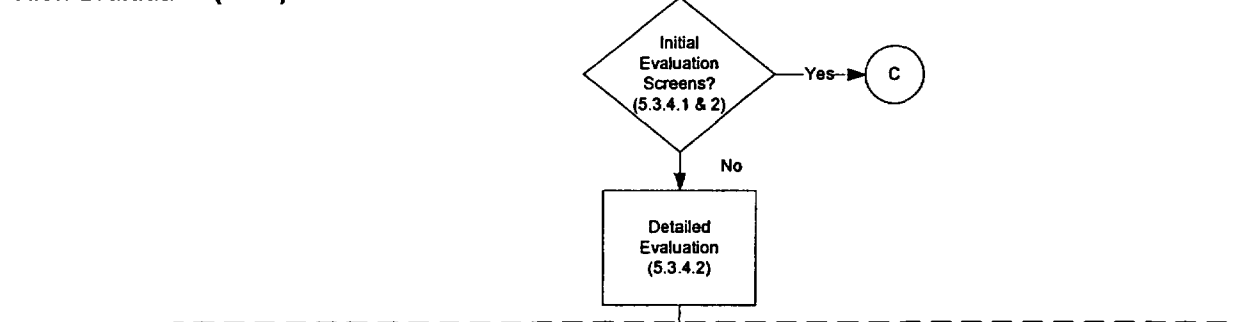
Defining the Change (5.3.2)



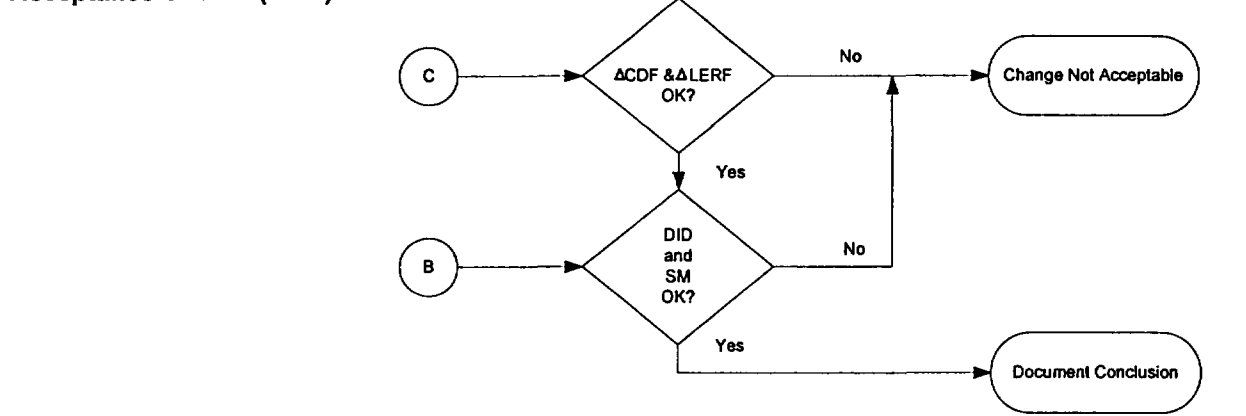
Preliminary Risk Screening (5.3.3)



Risk Evaluation (5.3.4)



Acceptance Criteria (5.3.5)



NEI 04-02 Figure 5-1 Revision 1



**REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR
REACTOR SAFEGUARDS
October 6, 2005**

**Sunil Weerakkody, Chief
Fire Protection Section
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
October 6, 2005**

OBJECTIVE

- Inform ACRS about how the NRC Staff and NEI addressed ACRS Comments
- Inform ACRS about other issues remaining to be addressed
- Seek ACRS agreement with respect to changes made to the Regulatory Guide and NEI-04-02 to address ACRS comments.

STAFF INTRODUCTION

■ Outline:

- Status of NFPA 805 Implementation – Paul Lain
- Changes to the Regulatory Guide and NEI-04-02 – Robert Radlinski
- Additional changes to the Regulatory Guide – Sunil Weerakkody

NEXT STEP

- Will provide the finalized Regulatory Guide and NEI-04-02 to ACRS, and seek endorsement to issue the Regulatory Guide after all changes are made



**REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR
REACTOR SAFEGUARDS
October 6, 2005**

**Paul Lain. Project Manager for NFPA 805 Implementation
Fire Protection Section
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001**

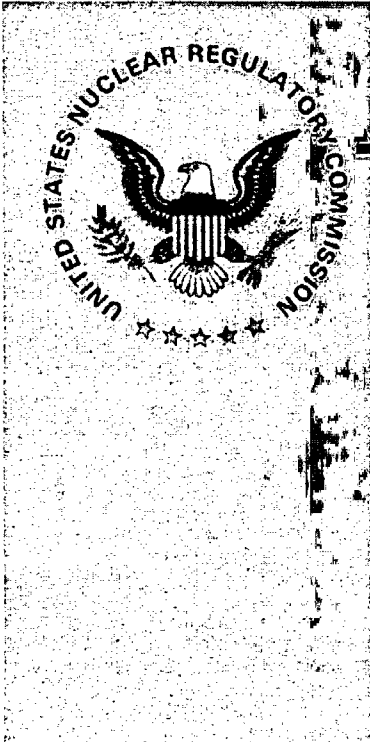
Status of NFPA 805 Implementation

- Duke Power Sends Letter of Intent – 2/28/05
(3 sites, 7 plants)
- Progress Energy Sends Energy Letter of Intent –
6/10/05
(4 sites, 5 plants)
- NRC/RGN II/Progress Energy/Duke Power Kicks Off
of Pilot Implementation – 8/11/05
- First Observation Visit Scheduled for November 2005

High Level Issues on Transition Plans

- Use of Fire PRAs

- Each plant that transitions to NFPA 805 plans to trace cables and develop or enhance Fire PRAs.
- Progress Energy requested an extension to the discretion period to develop Fire PRAs.
- DSSA and Office of Enforcement are considering changes to accommodate additional changes necessary to enforcement discretion policy to enable the development of fire PRAs
- Staff has informed licensees that transition to NFPA 805 without a fire PRA is impractical
- Staff plans to use completed and emerging regulatory guides, RES products, and industry standards to ensure that the pilots rely on acceptable methods for PRA and fire Modeling



**REGULATORY GUIDE FOR NFPA 805 RULE
ADVISORY COMMITTEE FOR
REACTOR SAFEGUARDS
October 6, 2005**

**Bob Radlinski
Fire Protection Engineer
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001**

OBJECTIVE

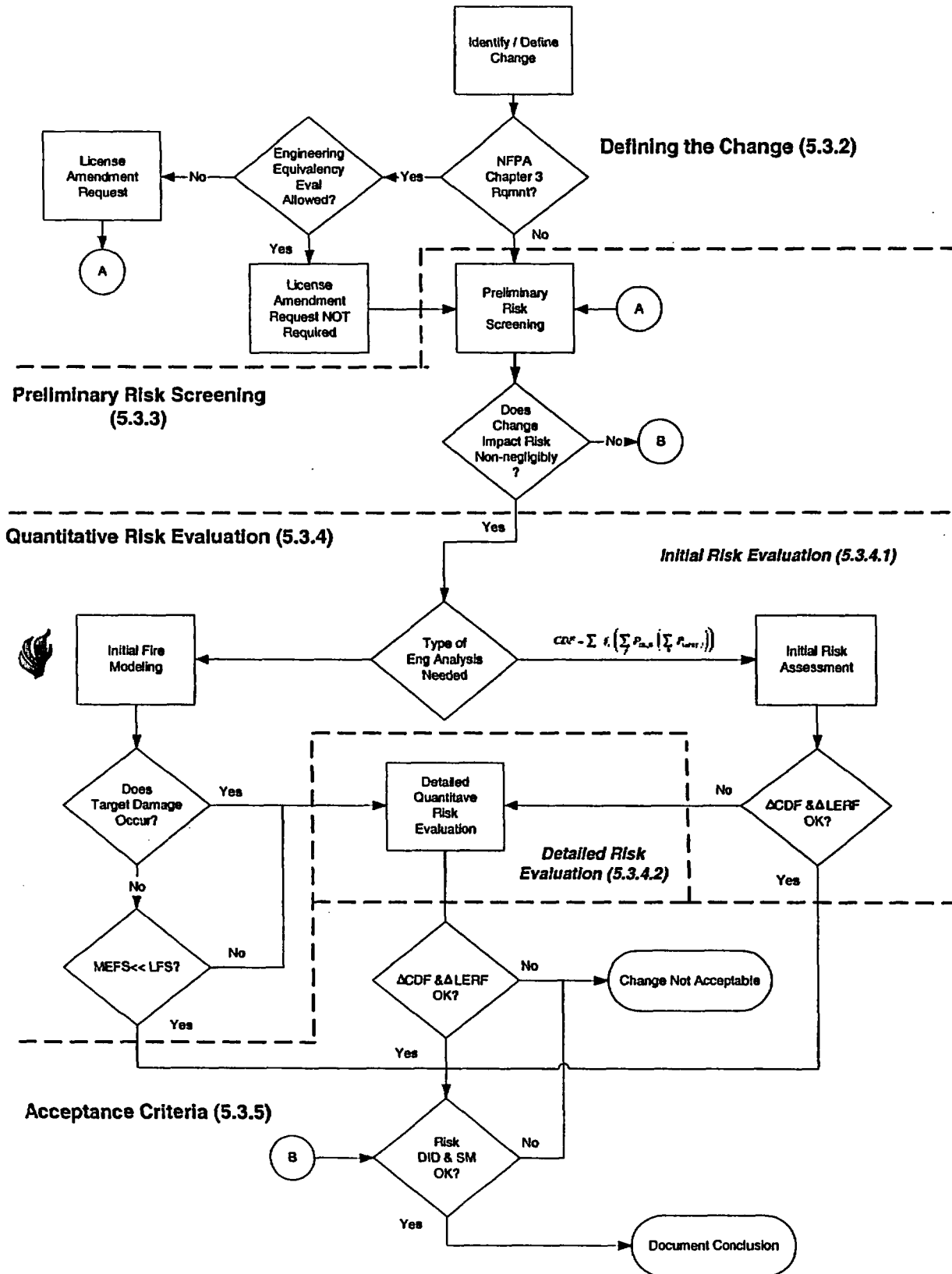
- Review ACRS comments and Staff responses
- Describe changes made to the NFPA 805 Regulatory Guide and NEI 04-02 to address ACRS comments

ACRS Comments

- **ACRS Comment:** The Regulatory Guide should not be issued in its present form.
- **How Addressed:** The Regulatory Guide has been revised to incorporate the ACRS comments. We plan to issue the Reg Guide next year after submitting draft final versions of the Reg Guide and NEI 04-02 to the ACRS in December 2005

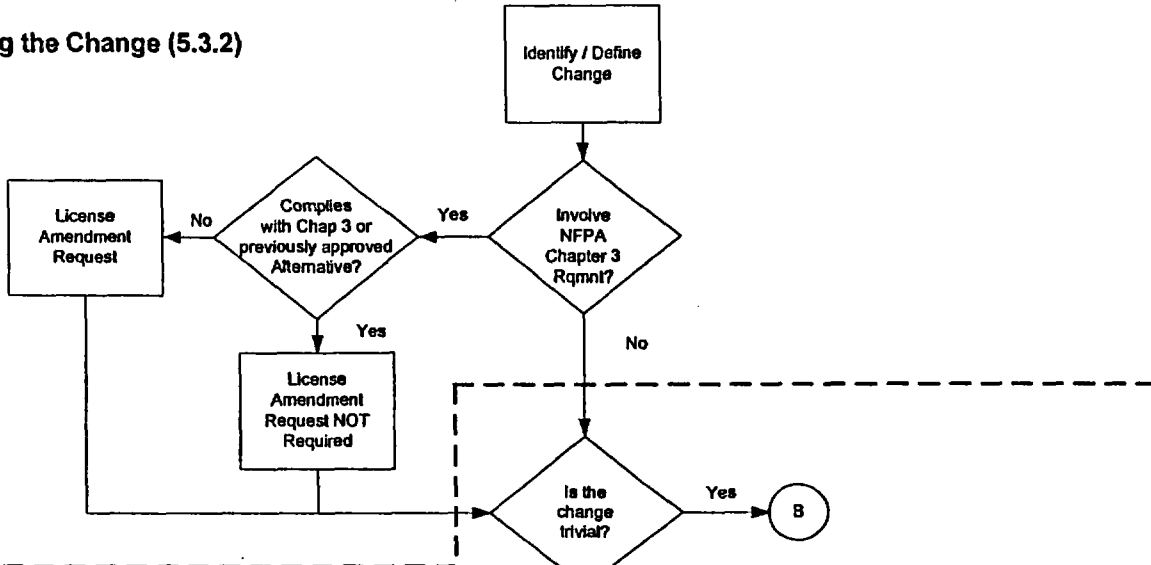
ACRS Comments (cont)

- ACRS Comment: The “initial fire modeling” approach should not be used as an alternative to estimates of changes in CDF and LERF.
- How addressed: NEI 04-02 has been revised to eliminate this approach (See revised Figure 5-1)

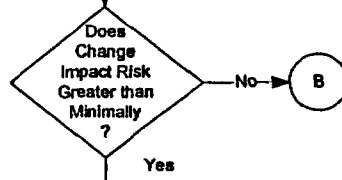


NEI 04-02 Figure 5-1 Revision 0

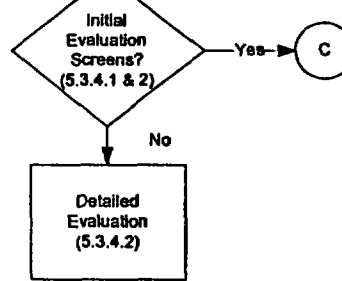
Defining the Change (5.3.2)



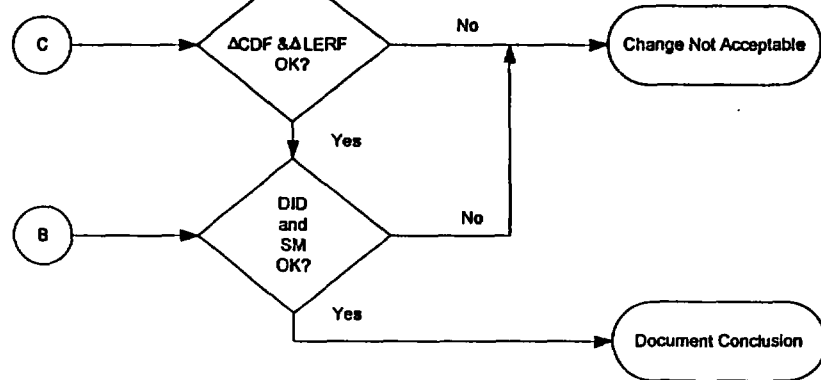
Preliminary Risk Screening (5.3.3)



Risk Evaluation (5.3.4)



Acceptance Criteria (5.3.5)



NEI 04-02 Figure 5-1 Revision 1

ACRS Comments (cont)

- ACRS Comment: The staff should not endorse methods for evaluating Δ CDF and Δ LERF that are not based on a fire PRA
- How Addressed: 10 CFR 50.48(c) and NFPA 805 allow risk assessments to be performed without a full fire PRA. However, to the extent possible, licensees are encouraged to develop a full fire PRA and the Regulatory Guide does not specifically endorse non-PRA methods.

ACRS Comments (cont)

- ACRS Comment: NEI 04 -02 contains many statements that are inconsistent with the Commission's policy of promoting the use of PRA methods. In the Regulatory Guide, the staff should make it clear that it does not endorse such statements.
- How Addressed: Statements in Appendix J, "Plant Change Evaluations", and Section 5.3, "Plant Change Process", were revised to be consistent with revised Figure 5 -1.

ACRS Comments (cont)

- **ACRS Comment:** The staff should ensure that the parts of NEI 04-02 that it endorses use correct methodology and language.
- **How Addressed:**
 - Held public meeting to share ACRS comments with NEI and discuss resolution
 - Held several follow-up phone calls with NEI
 - NRR and RES staff members reviewed all draft revisions to NEI 04-02
- Based on our reviews and discussions with NEI, we believe that the methodology and language used in the final version are correct.

NFPA 805 Regulatory Guide Changes

- Agreed with ACRS Comments and incorporated in final documents
- State that risk evaluations (for non-screened changes) should use PRA methods and tools
- Added PRA quality references including RG 1.174, RG 1.200 and the ANS fire PRA standard
- Noted that future additional guidance for fire PRAs will follow these referenced documents

NEI 04-02 Changes

- Agreed with ACRS comments and incorporated in final documents
- Eliminated all statements indicating that a change could be evaluated using the fire modeling approach without a risk assessment
- Encourages licensees to use a detailed, quantitative approach to plant change risk assessments

NEI 04-02 Changes (cont)

- Clarifies safety factors used to address uncertainties associated with fire models (the Reg Guide includes statement that margin must be large enough to bound uncertainties)
- Clarifies simplified assessment of Δ LERF

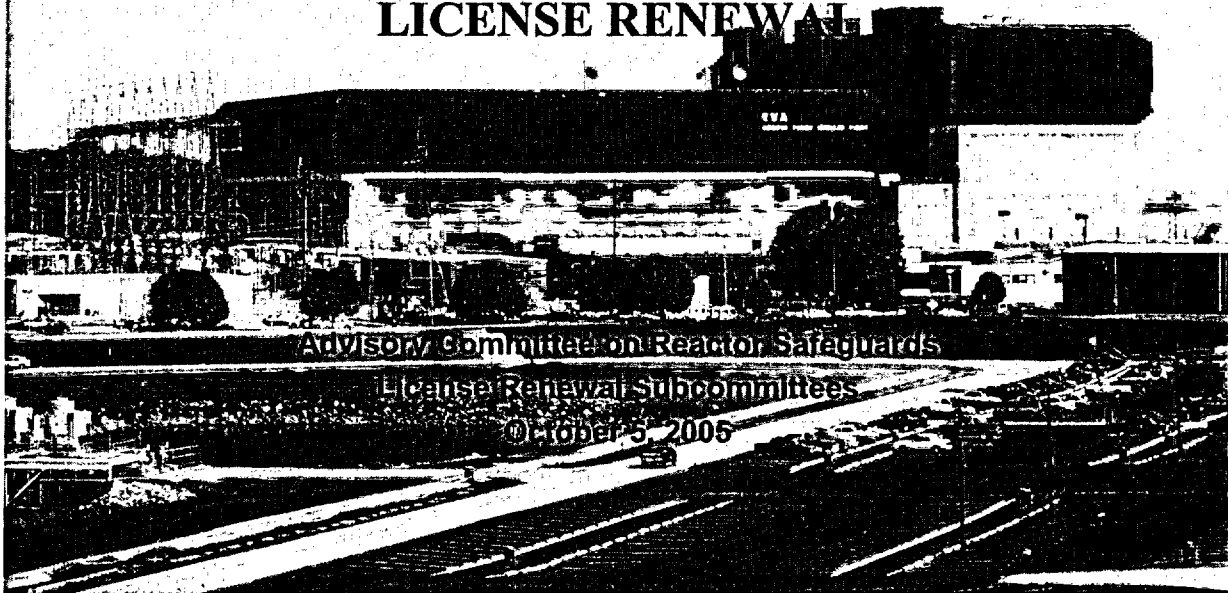
OTHER ISSUES

- 10 CFR 50.69, 10 CFR 50.46(a) & 10 CFR 50.48(c)
- “Self-Approval”
 - Prior approval of methods
 - Threshold values
- Cumulative Risk
 - Baseline CDF
 - Tracking

TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT



LICENSE RENEWAL



Agenda



- Description of Browns Ferry
- BFN License Renewal Application
 - Scoping
 - Time-Limited Aging Analysis
 - Aging Management Programs
- Unit 1 Layup
- Unit 1 Operating Experience
- License Renewal Commitments
- Open Items

Description of Browns Ferry



- All Three BFN Units are General Electric BWR 4 Reactors with Mark I Containments
- Approximate Years of Operation
 - Unit 1 – 10
 - Unit 2 – 23
 - Unit 3 – 18
- BFN Units 2 and 3 in Operation Since Recovery in 1991 and 1995, Respectively
- Unit 1 in Recovery Outage with Restart Scheduled for May 2007
 - Unit 1 will be operationally identical to Units 2 and 3
- NRC Performance Indicators Green

2

BFN License Renewal Application



- Three-Unit Application Submitted December 31, 2003
- Original License Expiration
 - Unit 1 – December 20, 2013
 - Unit 2 – June 28, 2014
 - Unit 3 – July 2, 2016
- License Renewal Application at Current Licensed Thermal Power for each Unit (Unit 1 – 3293 MWt, Units 2 and 3 – 3458 MWt)
- Appendix F Describes the Differences Between Unit 1 and Units 2 and 3
 - These differences will be eliminated prior to Unit 1 restart (May 2007)
- Requests for Additional Information ~230 (13 are Environmental, Remainder are Safety Evaluation)

3

Scoping



- **Scoping Basis**
 - Updated Final Safety Analysis Report
 - Safe Shutdown Analysis calculation
 - Maintenance Rule documentation
 - Controlled Plant Component Database
 - Licensing Basis and Design Basis documents
- **Specific Scoping for Regulated Events**
 - Fire Protection
 - Environmental Qualification
 - Anticipated Transients Without Scram
 - Station Blackout
- **77 Mechanical / Electrical Systems in Scope**

4

Time-Limited Aging Analysis



- **Neutron Embrittlement of the Reactor Vessel and Internals**
- **Metal Fatigue**
- **Environmental Qualification of Electrical Equipment**
- **Primary Containment Fatigue**
- **Plant Specific Time-Limited Aging Analyses**
 - Reactor Building Crane Load Cycles
 - Radiation Degradation of Drywell Expansion Gap Foam
 - Irradiation Assisted Stress Corrosion Cracking (IASCC) of Reactor Vessel Internals
 - Stress Relaxation of the Core Plate Hold-Down Bolts
 - Emergency Equipment Cooling Water Weld Flaw Evaluation

5

Reactor Vessel Time-Limited Aging Analysis



- Neutron Embrittlement of the Reactor Vessel and Internals
 - Unit 1: Conservatively evaluated using 54 Effective Full Power Years at Extended Power Uprate conditions
 - Peak fluence of limiting weld 1.95×10^{18} n/cm²
 - Units 2 and 3: Conservatively evaluated using 52 Effective Full Power Years at Extended Power Uprate conditions
 - Peak fluence of limiting weld 2.3×10^{18} n/cm²

6

License Renewal Aging Management Programs



- 39 Aging Management Programs Total
 - 38 are common to Units 1, 2, and 3
 - 1 is for only Unit 1 (i.e., Unit 1 Periodic Inspection Program)
- 11 Existing Aging Management Programs Requiring No Enhancement
- 11 Existing Aging Management Programs Revised Only to Include Unit 1
- 11 Existing Aging Management Programs Require Enhancement for all Units
- 6 New Aging Management Programs

7

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring No Enhancement
 - 10 CFR 50 Appendix J Program
 - Above ground Carbon Steel Tanks Program
 - ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program
 - ASME Section XI Subsection IWE Program
 - Bolting Integrity Program
 - BWR Control Rod Drive Return Line Nozzle Program
 - Diesel Starting Air Program
 - Fuel Oil Chemistry Program
 - Inspection of Overhead Heavy Load and Light Load Handling Systems Program
 - Reactor Head Closure Studs Program
 - Systems Monitoring Program

8

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring Revision Only to Incorporate Unit 1
 - BWR Feedwater Nozzle Program
 - BWR Penetrations Program
 - BWR Reactor Water Cleanup System Program
 - BWR Stress Corrosion Cracking Program
 - BWR Vessel Inside Diameter Attachment Welds Program
 - Chemistry Control Program
 - Closed-Cycle Cooling Water System Program
 - Environmental Qualification Program
 - Fire Protection Program
 - Flow-Accelerated Corrosion Program
 - Open-Cycle Cooling Water System Program

9

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring Enhancement (All Units)
 - ASME Section XI Subsection IWF Program
 - Buried Piping and Tanks Inspection Program
 - BWR Vessel Internals Program (includes steam dryers)
 - Compressed Air Monitoring Program
 - Electrical cables not subject to 10 CFR 50.49 Environmental Qualification requirements used in Instrumentation Circuits Program
 - Fatigue Monitoring Program
 - Fire Water System Program
 - Inspection of Water-Control Structures Program
 - Masonry Wall Program
 - Reactor Vessel Surveillance Program
 - Structures Monitoring Program

10

License Renewal Aging Management Programs



- New Aging Management Programs (for all Three Units)
 - Accessible Non-Environmental Qualification Cables and Connections Inspection Program
 - Bus Inspection Program
 - Inaccessible medium voltage cables not subject to 10 CFR 50.49 Environmental Qualification Requirements Program
 - One-Time Inspection Program
 - Selective Leaching of Materials Program
- New Aging Management Program (Unit 1 Only)
 - Unit 1 Periodic Inspection Program

11

One-Time Inspection Program



- Applies to Units 1, 2, and 3
- Verifies the Effectiveness of Aging Management Programs by Confirming that Unacceptable Degradation is not Occurring
- Where No Aging Management Program is Identified, the Inspections Confirm either:
 - Aging Effects are not Occurring, or
 - Aging Effects are Occurring at a Rate that does not Affect the Intended Function
- To be Completed Prior to the Period of Extended Operation
- Examples of inspection items:
 - Reactor coolant pressure boundary piping, valves, fittings less than four inches
 - Bottom thickness of above ground tanks
 - Submerged concrete and component supports
 - Ventilation ducts

12

Unit 1 Periodic Inspection Program



- Periodic Inspections will be Performed after Unit 1 is Returned to Operation to Verify No Additional Aging Effects are Occurring
- The Periodic Inspection Sample Locations will be a Subset of Non-Replaced Piping Locations (takes credit for restart inspections)
- First Round of Periodic Inspections will be Completed Prior to Period of Extended Operation and after Several Years of Unit 1 Operation
- An Inspection will be Performed during the Period of Extended Operation
- Subsequent Inspection Frequency will be Determined Based on Inspection Results

13

Unit 1 Layup Program



- **Criteria**
 - EPRI NP-5106, "Sourcebook for Plant Layup and Equipment Preservation", Revisions 0 (1987) and 1 (1992)
- **Types of Layup**
 - Dry
 - Wet
- **Lessons Learned from Unit 3 Layup and Subsequent Restart Applied to Unit 1**

14

Unit 1 Layup Program



- **Examples of Systems in Layup**
 - **Dry**

Core Spray	Condensate
Reactor Core Isolation Cooling	Feedwater
High Pressure Coolant Injection	Off Gas
Residual Heat Removal	Main Steam
 - **Wet**

Reactor Vessel	Control Rod Drive
Recirculation	
- **Results Met or Exceeded EPRI Guidelines**
- **Performed Visual, Surface, Ultrasonic, and Remote Inspections to Assess Unit 1 Condition**
- **No credit was taken for the lay-up program in determining the acceptability of structures, systems, or components for Unit 1 restart**

15

Unit 1 Operating Experience



- 10 CFR 54.17(c):
An application for a renewed license may not be submitted to the Commission earlier than 20 years before the expiration of the operating license currently in effect.
- Unit 1 Met this Requirement
- Unit 2 and Unit 3 Operating Experience is Applicable to Unit 1

16

Unit 1 Operating Experience



- Unit 1 has 10 Years of Operation
- Unit 3 Shutdown for 10 Years
 - Extensive layup experience with Unit 3 directly applicable to Unit 1
 - No layup induced aging effects during 10 years of ensuing operation
- Layup Experience from Unit 3 Incorporated into Unit 1 Recovery
 - RHR service water piping
 - Small bore piping
- Unit 1's Licensing Basis will be the same as that of Unit 2 and Unit 3 at Restart (Appendix F)
- Unit 1's Design, Configuration, Operating Procedures, Technical Specifications, and Updated Final Safety Analysis Report Identical to Unit 2 and Unit 3
- Internal and External Plant Operating Experience Incorporated into BFN Corrective Action Program

17

License Renewal Commitments



- Commitments made Through Application and Requests for Additional Information
- Tracked with Onsite Commitment Tracking System and Corrective Action Program
- ~ 114 Commitments made to Date

18

Open Items



- Core Plate Hold-Down Bolts
- Drywell Shell Corrosion
- Inspection of RHRSW Piping

19

Summary



- **Three Unit Application at Current Licensed Thermal Power**
- **Prepared using Generic Aging Lessons Learned Report (Rev. 0, 2001)**
- **Appendix F ensures Unit 1 Differences are resolved prior to Restart of the Unit**
- **Unit 2 and Unit 3 Operating Experience is Applicable to Unit 1**



*United States
Nuclear Regulatory Commission*

GENERIC ISSUE 80 PIPE BREAK EFFECTS ON CRD HYDRAULIC LINES IN BWRs

**Presented by
Abdul Sheikh
Harold VanderMolen
Office of Nuclear Regulatory Research**

October 6, 2005

1

Safety Significance

- Initiating event is a large break LOCA
- If pipe break is near CRD hydraulic lines, whipping pipe may crimp or kink some withdraw lines, preventing a cluster of rods from scrambling
- ECCS refills reactor vessel with cold water
- Possible reactivity excursion
- Additional post-LOCA heat source

2

Findings

- Core damage frequency well below thresholds
- Public risk well below thresholds

3

History of GI-80

- Raised by ACRS in 1983
- Prioritized as low priority in 1984, based on pipe layout geometry
- Closed out in 1995
- Reopened in 1998, based on discovery of new piping configurations
- NUREG/CR-6395 in 1999 identified breaks in RCS and RHR piping that may damage CRD piping in BWR plants

4

Technical Assessment Objective

- Perform detailed analysis of RHR and RCS break interactions with CRD piping
- Determine if complete closure of CRD piping on impact with RHR and RCS piping is possible
- Determine revised CDF based on a conservative estimate of the probability of crimping of CRD piping
- Use revised CDF to determine appropriate rating for GI-80 in accordance with MD 6.4

5

BWR Mark I and Mark II RCS Piping and CRD Arrangements

- The RHR and RCS piping layout and configuration for GE Mark I and Mark II is essentially the same
- The CRD piping layout for Mark I GE2 is different from Mark I/II GE3/GE4/GE5
- Mark I GE 2 has three sets of CRD bundles
- Mark I/II GE3/GE4/GE5 has four sets of CRD bundles

6

Mark I/II GE3/GE4/GE5 CRD Piping Layout



9

Event Quantification Process



CDF for RHR/RCS piping impact on CRD piping is a function of the following factors

- Pipe rupture initiating event (IE)
- Fraction of piping considered in IE that is from RHR or RCS system (PIPETYPE)
- Fraction of RHR or RCS system piping that can impact CRD piping (TYPEFRAC)
- Probability of pipe whip or jet impingement that can cause CRD system failure (RUPTPROB)

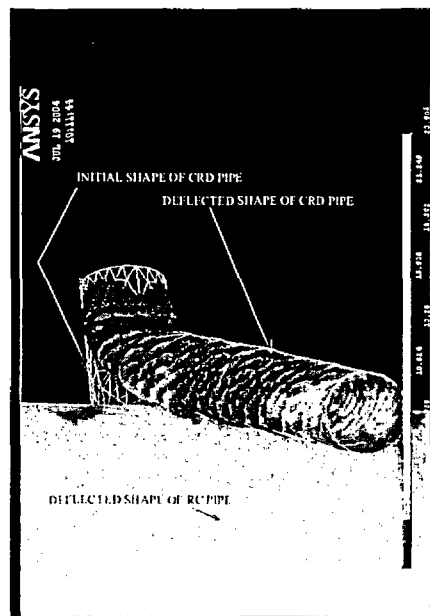
10

Probability of CRD System Failure (RUPTPROB) for Mark I-GE2 Containment

- RHR piping cannot impact CRD piping
- CRD piping bundle located 18 feet above the postulated break location of RCS piping
- 25 inch clearance between RCS and CRD piping
- RCS piping will fail (plastic hinge will form) before it can impact the CRD bundle
- Finite element analysis performed for a hypothetical impact to CRD piping
- CRD piping flexible and will bend without significant crushing or crimping before rupture
- Analysis results consistent with the behavior in the test results documented in NUREG/CR-3231
- RUPTPROB can be conservatively taken as 0.1 for CDF calculations

11

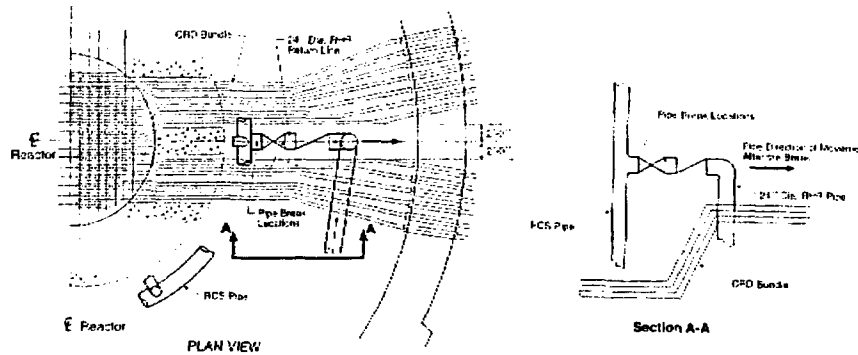
Deflected Shape of CRD Pipe



12

RUPTPROB for Mark I and II-GE3/GE4 Containment

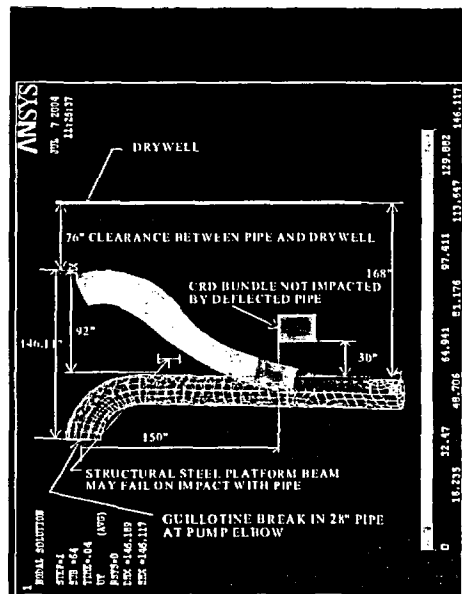
Impact of RHR pipe on the CRD piping is improbable
 RUPTPROB can be conservatively taken as 0.1 for CDF calculations



13

RUPTPROB for Mark I-GE3/GE4 Containment (Contd)

- Deflected Shape of the RCS Pipe Relative to CRD Piping
- Impact of RCS pipe on the CRD piping is improbable
- RUPTPROB can be conservatively taken as 0.1 for CDF calculations



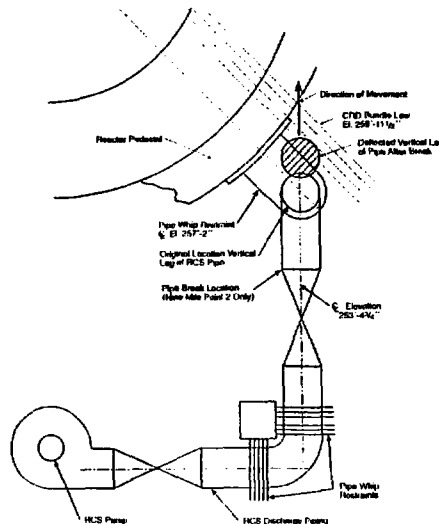
14

RUPTPROB for Mark II-GE5 Containment

- Layout of RHR and CRD piping similar to Mark II GE3/GE4 plants
- Pipe whip restraint will prevent RHR pipe to impact CRD piping
- There is a 18 inch gap between RHR and CRD piping
- RCS pump discharge pipe break downstream of isolation wall postulated only for the Nine Mile Point 2
- There is a possibility of RCS piping impacting the CRD bundle if the pipe whip restraint fails
- After impact with RCS piping, the CRD piping will bend without significant crushing or crimping before rupture
- Pipe whip restraints on the vertical leg of RCS piping and on circular header will prevent RCS pump vertical discharge pipe break impact on the CRD piping
- RUPTPROB can be conservatively taken as 0.1 for all CDF calculations

15

RCS Pump Discharge Pipe Break Downstream of Isolation Valve in Mark II-GE5 Containment



16

Probabilistic Analysis Approach

- Initiating event frequency – used “classic” large LOCA frequency, lognormal distribution
- PIPETYPE – Normal distribution, based on four existing licensee submittals
- TYPEFRAC – Normal distribution, based on review of plant drawings
- RUPTPROB – Exponential distribution, based on ANSYS calculations

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Probabilistic Analysis Results - Sequence frequency

Product Line	Sequence	Point Estimate	Mean	Median	5 th Percentile	95 th Percentile
GE2 Mark I	RCS	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
GE3&4 Mark I	RCS	3.9E-7	4.0E-7	8.6E-8	2.6E-9	1.6E-6
	RHR	1.3E-7	1.3E-7	2.3E-8	8.6E-10	5.4E-7
	total	5.2E-7	5.3E-7	1.2E-7	4.0E-9	2.1E-6
GE4 Mark II	RCS	3.0E-8	5.1E-8	8.3E-9	1.7E-10	2.0E-7
	RHR	2.7E-7	2.8E-7	5.9E-8	1.8E-9	1.1E-6
	total	3.0E-7	3.3E-7	7.2E-8	2.3E-9	1.3E-6
GE5 Mark II	RCS	2.7E-7	2.8E-7	5.9E-8	1.1E-9	1.1E-6
	RHR	3.0E-7	3.1E-7	6.6E-8	2.1E-9	1.3E-6
	total	5.8E-7	5.9E-7	1.3E-7	4.6E-9	2.4E-6

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Public Risk

- Estimated for GE4 in Mark I containment (most common)
- Based on NUREG-1150 plant damage state for ATWS initiated by stuck-open safety/relief valve
- Result was less than one person-rem per reactor-year

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Conclusions

- Core damage frequency and public risk are well below thresholds
- Generic Issue GI-80 will be closed out with no additional requirements

20

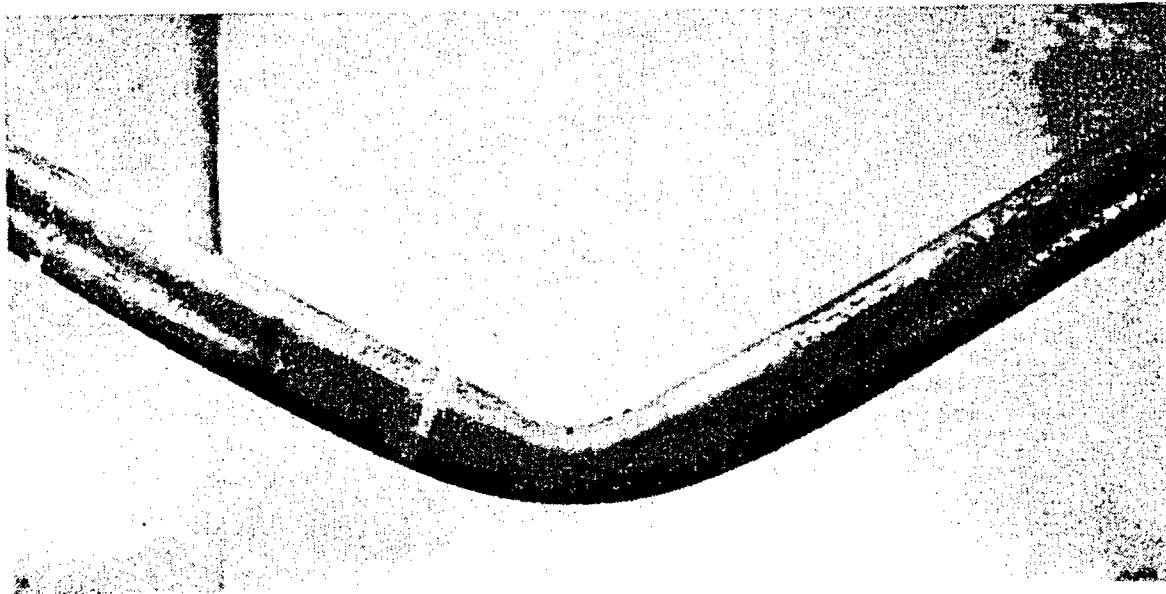


FIGURE 2.20. Impacted 3" SCH 160 Pipe

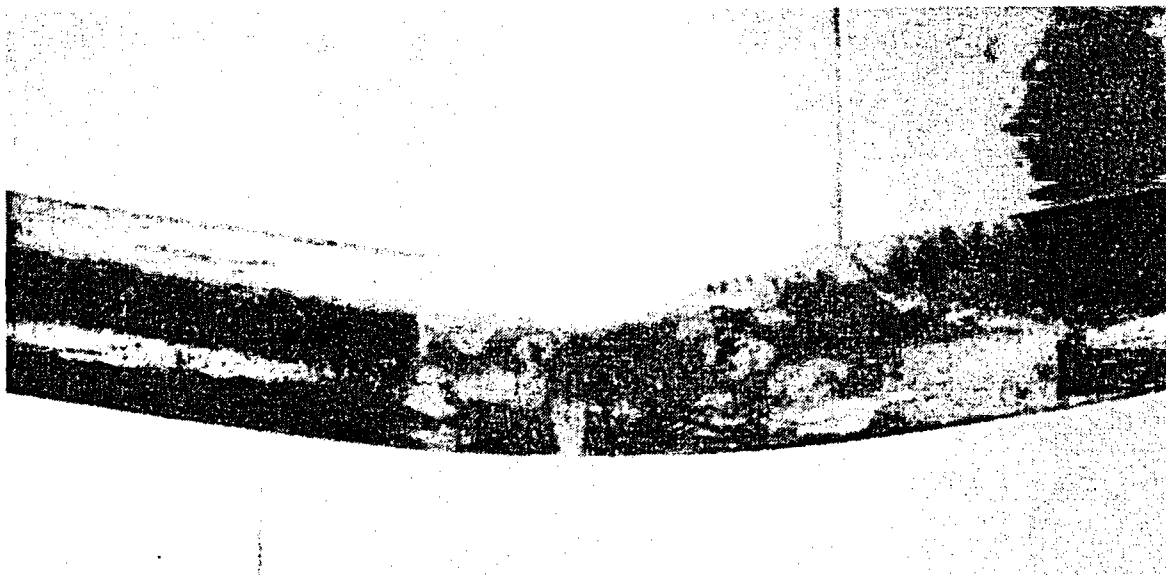


FIGURE 2.21. Impacted 6" SCH 40 Pipe

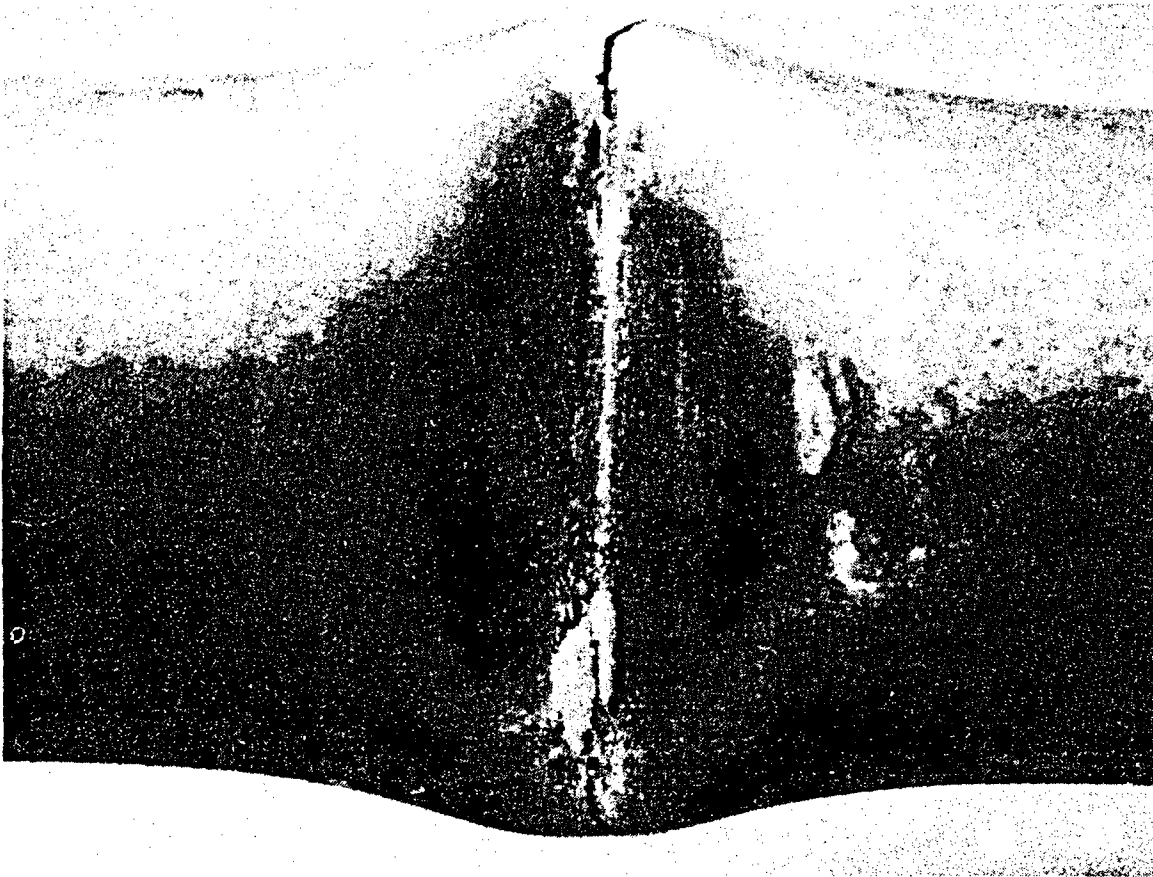


FIGURE 2.24. Failed 6" SCH 40 Pipe - Top View



FIGURE 2.25. Failed 6" SCH 40 Pipe - Side View

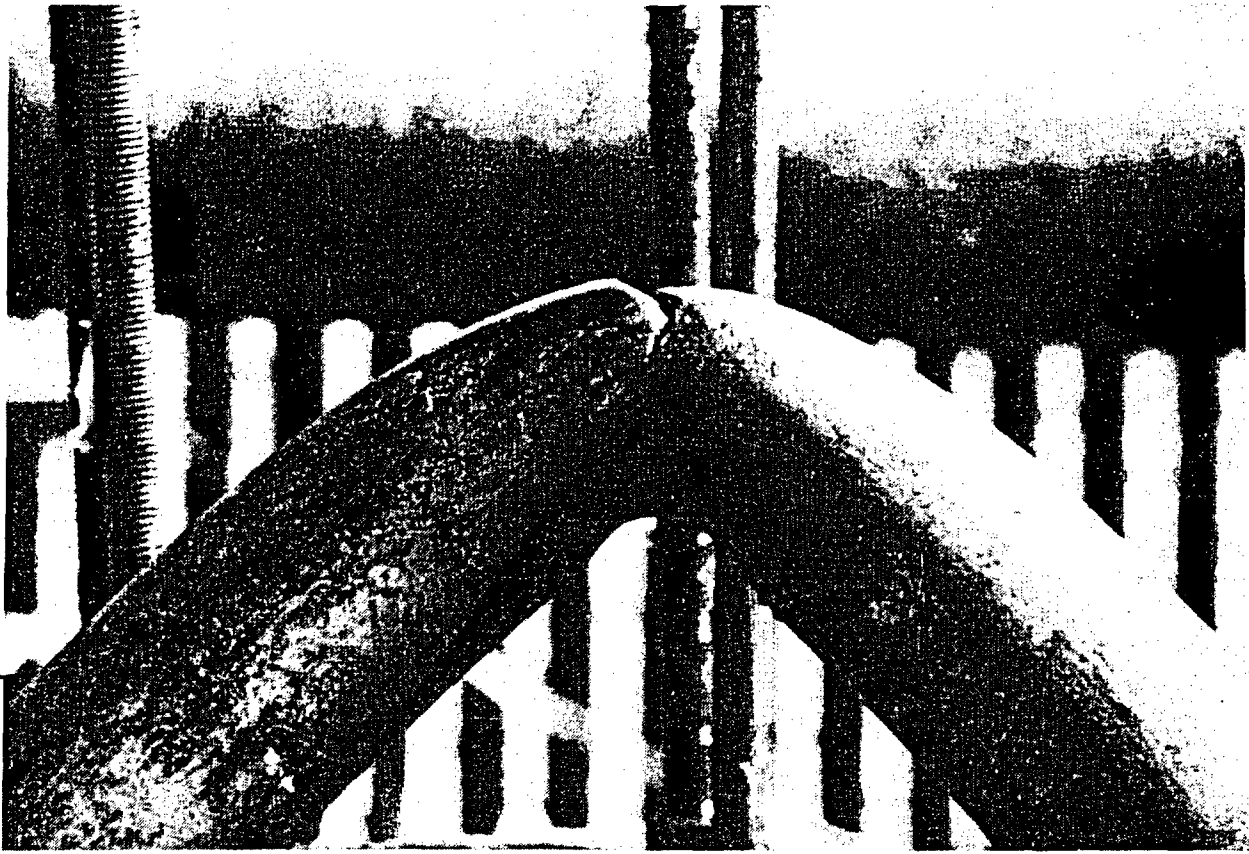


FIGURE 2.26. Failed 3" SCH 160 Pipe



Browns Ferry Nuclear Plant Units 1, 2, and 3 License Renewal Safety Evaluation Report

**Staff Presentation to the ACRS Full Committee
Ram Subbaratnam, and
Yoira Diaz Sanabria, Project Managers
Office of Nuclear Reactor Regulation
October 6, 2005**



Review Highlights

- **License extension request – December 31, 2003**
 - Unit 1: December 20, 2013
 - Unit 2: June 28, 2014
 - Unit 3: July 2, 2016
- **SER with Open and Confirmatory Items issued on August 9, 2005**
- **Four (4) Open Items**
 - 1 Scoping and Screening: OI 2.4-3
 - 1 Unit 1 Periodic Inspection: OI 3.0-3 LP
 - 1 Time-limited aging analysis: OI 4.7.7
 - 1 RHRSW Piping: Inspection Finding
- **Two (2) Confirmatory Items**
- **Four (4) License Conditions; Three Standard and one specific to BFN**
- **Fourth Condition requires completion of Unit 1 CLB differences (13 Items) described in LRA Appendix F**

Section 2.4: Scoping and Screening of Containments, Structures and Supports



Open Item 2.4-3 Drywell Shell Corrosion

- Potential Corrosion of the inaccessible portion of the Drywell affected by the leakage from the refueling cavity seal
- The staff proposed two options:
 - (1) Bring the refueling cavity seal in the scope of LR or
 - (2) Periodically monitor the potential degradation of the inaccessible side of the dry well



Section 3.7: AMR of Unit 1 Systems in Lay Up

Open Item 3.0-3 LP

- Unit 1 Periodic Inspection Program
 - BFN submitted Unit 1 Periodic Inspection Program
 - Staff in reviewing the Program element needed additional confirmation.
 - Section 3.0 of the final SER will include the staff evaluation of this program



Section 4.7.7: Stress Relaxation Core Plate Hold-Down Bolts

- TLAA evaluated for loss of pre-load in accordance with 10 CFR 54.21 (c)(1)(ii)
- Expected loss of preload of 20% which bounds the original BWRVIP-25 value
- In a plant specific analysis
 - Core plate hold-down bolts will maintain sufficient preload to prevent sliding of core plate
 - Hold-down bolts meet ASME Section III, Class 1, Level D service limits at the end of PEO

Section 4.7.7: Stress Relaxation Core Plate Hold-Down Bolts



- **Open Item 4.7.7**
 - The staff reviewed the method of analysis based on GE's generic stress relaxation data on irradiated stainless steel materials

 - The staff requested additional information to address
 - Horizontal and vertical loads for all operating conditions during PEO
 - Sliding of core plate from core plate rim during the PEO
 - Axial and bending stresses of bolts



**Browns Ferry Nuclear Plant
Units 1, 2, and 3
License Renewal
Safety Evaluation Report**

**Caudle Julian, Senior Project Manger
DRP, Region II**

Browns Ferry License Renewal Inspection



- AMP inspection conducted November 29 - December 17, 2004
- Inspection concluded that existing programs to be credited as aging management programs for license renewal are generally functioning well.
- Inspectors observed that the applicant had not yet begun the implementation process for new and enhanced AMPs
- AMP procedures have yet to be defined and composed

Browns Ferry License Renewal Inspection



- For existing programs, the identification and selection of which particular existing procedures constitute the AMP had yet to be done
- Region II concluded that NRC will perform another inspection when the applicant has progressed further with AMP implementation.
- In walking down plant systems and examining plant equipment the inspectors found no significant adverse conditions and it appears plant equipment was being maintained adequately.

Browns Ferry License Renewal Inspection



Second (optional) AMP Inspection

- Conducted September 19 - 23, 2005
- Reviewed sample of 40 AMP Implementation Packages containing proposed procedures
 - Packages contained some errors and were not meticulously reviewed
 - Applicant initiated PER for corrective action

Browns Ferry License Renewal Inspection



Second (optional) AMP Inspection (cont.)

- Reviewed plans for tracking future actions using TROI system
 - Not initially linked to Implementation Packages but quickly corrected
 - Inspection sample commitments were included
 - Much duplication and varying format resulting in confusing document
- Applicant decided to track future actions using PER process
- Region II will follow up on these issues during a future inspection



Conclusion

- Region II concluded that NRC will perform another inspection when the applicant has progressed further with AMP implementation.
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**Caudle Julian, Senior Project Manger
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7

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12

TENNESSEE VALLEY AUTHORITY BROWNS FERRY NUCLEAR PLANT



LICENSE RENEWAL

Advisory Committee on Reactor Safeguards

License Renewal Subcommittees

October 5, 2005





Agenda

- Description of Browns Ferry
- BFN License Renewal Application
 - Scoping
 - Time-Limited Aging Analysis
 - Aging Management Programs
- Unit 1 Layup
- Unit 1 Operating Experience
- License Renewal Commitments
- Open Items



Description of Browns Ferry

- All Three BFN Units are General Electric BWR 4 Reactors with Mark I Containments
- Approximate Years of Operation
 - Unit 1 – 10
 - Unit 2 – 23
 - Unit 3 – 18
- BFN Units 2 and 3 in Operation Since Recovery in 1991 and 1995, Respectively
- Unit 1 in Recovery Outage with Restart Scheduled for May 2007
 - Unit 1 will be operationally identical to Units 2 and 3
- NRC Performance Indicators Green

BFN License Renewal Application



- Three-Unit Application Submitted December 31, 2003
- Original License Expiration
 - Unit 1 – December 20, 2013
 - Unit 2 – June 28, 2014
 - Unit 3 – July 2, 2016
- License Renewal Application at Current Licensed Thermal Power for each Unit (Unit 1 – 3293 MWt, Units 2 and 3 – 3458 MWt)
- Appendix F Describes the Differences Between Unit 1 and Units 2 and 3
 - These differences will be eliminated prior to Unit 1 restart (May 2007)
- Requests for Additional Information ~230 (13 are Environmental, Remainder are Safety Evaluation)



Scoping

- Scoping Basis
 - Updated Final Safety Analysis Report
 - Safe Shutdown Analysis calculation
 - Maintenance Rule documentation
 - Controlled Plant Component Database
 - Licensing Basis and Design Basis documents
- Specific Scoping for Regulated Events
 - Fire Protection
 - Environmental Qualification
 - Anticipated Transients Without Scram
 - Station Blackout
- 77 Mechanical / Electrical Systems in Scope



Time-Limited Aging Analysis

- Neutron Embrittlement of the Reactor Vessel and Internals
- Metal Fatigue
- Environmental Qualification of Electrical Equipment
- Primary Containment Fatigue
- Plant Specific Time-Limited Aging Analyses
 - Reactor Building Crane Load Cycles
 - Radiation Degradation of Drywell Expansion Gap Foam
 - Irradiation Assisted Stress Corrosion Cracking (IASCC) of Reactor Vessel Internals
 - Stress Relaxation of the Core Plate Hold-Down Bolts
 - Emergency Equipment Cooling Water Weld Flaw Evaluation

Reactor Vessel Time-Limited Aging Analysis



- Neutron Embrittlement of the Reactor Vessel and Internals
 - Unit 1: Conservatively evaluated using 54 Effective Full Power Years at Extended Power Uprate conditions
 - Peak fluence of limiting weld 1.95×10^{18} n/cm²
 - Units 2 and 3: Conservatively evaluated using 52 Effective Full Power Years at Extended Power Uprate conditions
 - Peak fluence of limiting weld 2.3×10^{18} n/cm²

License Renewal Aging Management Programs



- 39 Aging Management Programs Total
 - 38 are common to Units 1, 2, and 3
 - 1 is for only Unit 1 (i.e., Unit 1 Periodic Inspection Program)
- 11 Existing Aging Management Programs Requiring No Enhancement
- 11 Existing Aging Management Programs Revised Only to Include Unit 1
- 11 Existing Aging Management Programs Require Enhancement for all Units
- 6 New Aging Management Programs

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring No Enhancement
 - 10 CFR 50 Appendix J Program
 - Above ground Carbon Steel Tanks Program
 - ASME Section XI Inservice Inspection Subsections IWB, IWC, and IWD Program
 - ASME Section XI Subsection IWE Program
 - Bolting Integrity Program
 - BWR Control Rod Drive Return Line Nozzle Program
 - Diesel Starting Air Program
 - Fuel Oil Chemistry Program
 - Inspection of Overhead Heavy Load and Light Load Handling Systems Program
 - Reactor Head Closure Studs Program
 - Systems Monitoring Program

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring Revision Only to Incorporate Unit 1
 - BWR Feedwater Nozzle Program
 - BWR Penetrations Program
 - BWR Reactor Water Cleanup System Program
 - BWR Stress Corrosion Cracking Program
 - BWR Vessel Inside Diameter Attachment Welds Program
 - Chemistry Control Program
 - Closed-Cycle Cooling Water System Program
 - Environmental Qualification Program
 - Fire Protection Program
 - Flow-Accelerated Corrosion Program
 - Open-Cycle Cooling Water System Program

License Renewal Aging Management Programs



- Existing Aging Management Programs Requiring Enhancement (All Units)
 - ASME Section XI Subsection IWF Program
 - Buried Piping and Tanks Inspection Program
 - BWR Vessel Internals Program (includes steam dryers)
 - Compressed Air Monitoring Program
 - Electrical cables not subject to 10 CFR 50.49 Environmental Qualification requirements used in Instrumentation Circuits Program
 - Fatigue Monitoring Program
 - Fire Water System Program
 - Inspection of Water-Control Structures Program
 - Masonry Wall Program
 - Reactor Vessel Surveillance Program
 - Structures Monitoring Program

License Renewal Aging Management Programs



- New Aging Management Programs (for all Three Units)
 - Accessible Non-Environmental Qualification Cables and Connections Inspection Program
 - Bus Inspection Program
 - Inaccessible medium voltage cables not subject to 10 CFR 50.49 Environmental Qualification Requirements Program
 - One-Time Inspection Program
 - Selective Leaching of Materials Program
- New Aging Management Program (Unit 1 Only)
 - Unit 1 Periodic Inspection Program



One-Time Inspection Program

- Applies to Units 1, 2, and 3
- Verifies the Effectiveness of Aging Management Programs by Confirming that Unacceptable Degradation is not Occurring
- Where No Aging Management Program is Identified, the Inspections Confirm either:
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Unit 1 Layup Program

- Criteria
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- Types of Layup
 - Dry
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- Tracked with Onsite Commitment Tracking System and Corrective Action Program
- ~ 114 Commitments made to Date



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- Drywell Shell Corrosion
- Inspection of RHRSW Piping



Summary

- Three Unit Application at Current Licensed Thermal Power
- Prepared using Generic Aging Lessons Learned Report (Rev. 0, 2001)
- Appendix F ensures Unit 1 Differences are resolved prior to Restart of the Unit
- Unit 2 and Unit 3 Operating Experience is Applicable to Unit 1