Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: Advisory Committee on Reactor Safeguards
Materials, Metallurgy and Reactor Fuels
Subcommittee Meeting: Open Session

Docket Number: (n/a)

Location: Rockville, Maryland

Date: Monday, June 17, 2013

Work Order No.: NRC-012 Pages 1-82

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2 NUCLEAR REGULATORY COMMISSION

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4 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

5 (ACRS)

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7 MATERIALS, METALLURGY AND REACTOR FUELS SUBCOMMITTEE

8 OPEN SESSION

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10 MONDAY

11 JUNE 17, 2013

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13 ROCKVILLE, MARYLAND

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15 The Subcommittee met at the Nuclear

16 Regulatory Commission, Two White Flint North, Room T2B1,

17 11545 Rockville Pike, at 1 p.m., J. SAM ARMIJO, Chairman,

18 presiding.

19 COMMITTEE MEMBERS:

20 J. SAM ARMIJO, Chairman

21 SANJOY BANERJEE, Member

22 JOY REMPE, Member

23 WILLIAM J. SHACK, Member

24 GORDON R. SKILLMAN, Member

25

1 INVITED EXPERT:

2 ROBERT BALLINGER

3 DESIGNATED FEDERAL OFFICIAL:

4 ZENA ABDULLAHI

5

6 NRC STAFF

7 PAUL CLIFFORD (NRR)

8 IAN PORTER (RES/FSCA)

9 ANDREW PROFFITT (NRR/SNPB)

10 PATRICK RAYNAUD (RES)

11 STRUART RICHARDS (RES/DSA)

12 JEFFREY SCHMIDT (NRO/DSRA)

13 JOHN VOGLEWEDE (RES/DSA)

14

15 OTHER ATTENDEES

16 NICOLE BRICHACEK (WESTINGHOUSE)

17 PAUL KERSTING (WESTINGHOUSE)

18 ALAN LEVIN (AREVA)

19 DAVID MITCHELL (WESTINGHOUSE)

20 THOMAS RODACK (WESTINGHOUSE)

21 WILLIAM SLAGLE (BECHTEL/B&W MPOWER)

22 GREG SWINDLEHURST (EPRI/GS NUCLEAR)

23 KURT WALTER (MITSBISHI NUCLEAR ENERGY SYSTEMS)

24

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2 T-A-B-L-E O-F C-O-N-T-E-N-T-S

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

2 (12:58 p.m.)

3 CHAIRMAN ARMIJO: Good afternoon. The
4 meeting will now come to order. This is a meeting of the
5 Materials Metallurgy and Fuels Subcommittee of the
6 Advisory Committee on Reactor Safeguards. I'm Sam
7 Armijo, chairman of the subcommittee.

8 ACRS members in attendance today are Sanjoy
9 Banerjee, Gordon Skillman, William Shack and Joy Rempe.
10 Also joining us today is Professor Ron Ballinger,
11 member-elect to the ACRS, and welcome aboard, Ron. He's
12 still going through some paperwork, so he can't
13 officially participate in full committee votes and
14 things like that, but he's welcome and will participate
15 in the subcommittee.

16 Zena Abdullahi is the designated federal
17 official for this meeting.

18 The subject of today's subcommittee meeting
19 is the review of NRR's study investigating the
20 vulnerability of PWR cores to pellet cladding
21 interaction stress corrosion cracking failures during
22 anticipated operation occurrences, AOOs.

23 The research staff will also brief us on the
24 Studsvik Cladding Integrity Program, this is commonly
25 called the SCIP program. It's an experimental study

1 funded by research. It's designed to investigate the
2 dominant failure mechanisms for light water reactor fuel
3 cladding during normal operation or anticipated
4 transients with focus on the effects of burnup on fuel
5 performance.
6 The SCIP program covers multiple failure
7 mechanisms such as pellet cladding mechanical
8 interaction; PCI, which is a stress corrosion mechanism;
9 hydrogen embrittlement; and delayed hydride cracking,
10 which is commonly labeled DHC.
11 The discussion in today's meeting is
12 focused on the susceptibility of PWRs to PCI stress
13 corrosion cracking failures during specific transients.
14 Therefore, the research staff will describe the insights
15 into PCI/SCC mechanisms gained in recent years and from
16 recently completed investigations in the SCIP program,
17 as well as related aspects of the experiment's plan for
18 future phases of the program.
19 The Studsvik Cladding Integrity Program
20 session of the meeting will be closed to the public to
21 protect material that is proprietary to the Organization
22 of Economic Cooperation and Development, OECD, and its
23 participating members who sponsored the Studsvik work.
24 This is a broad-base participation. I
25 believe all of the vendors are involved to some extent

1 or another, but many, many organizations participate in
2 this program. I asked the NRC staff to verify that only
3 individuals cleared for this session of the meeting with
4 a need to know are present in the room. We
5 have a telephone bridge line for the open sessions of the
6 meeting. To minimize disturbances the line will be kept
7 in a listen-in only mode. Zena will make sure that that
8 is the case.
9 As the meeting is being transcribed, I
10 request the participants in this meeting to use the
11 microphones located throughout the room when addressing
12 the subcommittee. Participants should first identify
13 themselves and speak with sufficient clarity and volume
14 so that they can be readily heard.
15 We'll now proceed with the meeting, and
16 before I call on Paul Clifford to start I'd like to make
17 a couple of additional remarks.
18 The history of this topic is quite
19 interesting. In 1979, the ACRS wrote a letter to the
20 staff saying the issue of PCI fuel failures on transients
21 should be investigated. The staff did work, and in the
22 early 1980s, 1984 specifically, a task force was put
23 together consisting of Bob Van Houten, Mike Tokar, and
24 P.E. McDonald, I think it was Phillip.
25 And they prepared a document called PCI

1 Related Cladding Failures During Off-Normal Events. I
2 only have a draft version. I don't know if the NUREG was
3 ever actually issued. But it's a very interesting
4 thing.
5 But back that 1984 their conclusion was, "In
6 summary this task force is concluded, but there is a
7 reasonable chance that PCI failures will occur during
8 some off-normal reactor operating conditions." And
9 then they went on to say that there just wasn't that much
10 data to really resolve the problem.
11 Concurrent with that in the case of the
12 BWRs, a lot of work was done to develop a fuel cladding
13 that was very resistant to PCI, which actually was
14 resistant even in the AOO range. So the BWR issue is
15 probably -- and I'm using the word "probably" in quotes
16 -- okay. The thing that hadn't been investigated
17 sufficiently, and that was the subject of several
18 letters, more recent letters from the ACRS, is the PWRs.
19 And so the staff today will tell us what
20 they've done to try and analyze the PCI risk for PWRs
21 during AOOs, and I think it's going to be a very
22 interesting discussion. As far as I know, it's the first
23 quantitative analysis of the issue. So we're moving a
24 little slowly from 1979 to now, but it's government work
25 so we probably are okay.

1 With that I'll turn it over to Paul. You
2 can also rebut what I had to say.
3 MEMBER REMPE: Mr. Chairman, before you do
4 that so appropriate waivers are prepared, I need to
5 acknowledge that I may have some organizational conflict
6 of interest issues, and so I may need to limit my
7 participation in certain aspects of the discussion.
8 CHAIRMAN ARMIJO: Okay, thank you. Yes,
9 in cases like that if a member is free to ask questions
10 and contribute information, facts and things like that
11 it's a very remote possibility of conflict of interest,
12 but I'm glad you raised it.
13 Okay, Paul. Now we can get started. We
14 only took six minutes.
15 MR. CLIFFORD: Okay, good afternoon. I'll
16 start by walking through the agenda so we can kind of set
17 the stage here. I'll begin by providing background
18 information on PCI and PCMI and addressing the current
19 regulatory practice with respect to ensuring or
20 confirming control of fuel cladding integrity during
21 AOOs.
22 I'll be followed by my colleague Patrick
23 Reynaud from the Office of Research, and he will
24 summarize the valuable work that's being done as part of
25 SCIP on PCI.

1 Then I will walk through kind of the
2 operating conditions of a PWR, the LCOs, the co-operating
3 limits, and the reactor protection system functions in
4 a PWR, and we will walk through a survey of all of the
5 AOOs to determine which ones may be susceptible to PCI.
6 And then we will compare the power and
7 overpower conditions of those AOOs using FRAPCON to
8 calculate local cladding stress, and compare that
9 against an empirically derived failure threshold, and
10 then we'll finish up with some conclusions.
11 Okay, PCMI versus PCI. PCMI is simply the
12 mechanical interaction that occurs between the fuel
13 pellet and the clad ID. Fuel rods are manufactured with
14 a gap between the pellet and the cladding ID. It's about
15 7 mils. That's the standard number.
16 With burnup you have a radiation-induced
17 swelling, fuel pellet thermal expansion and cladding
18 creep-down results in reduction of this gap size.
19 Eventually the gap will close and there will be hard
20 contact. That occurs anywhere between 25 and 30
21 gigawatt-days before you'll have hard contact. So
22 that's PCMI. Now it's simply a mechanical interaction
23 stress and strain on the cladding.
24 Now in concert with an aggressive chemical
25 environment such as can be found in fission products,

1 specifically iodine, there can be a chemical interaction
2 in addition to the mechanical interaction, and that's
3 commonly referred to as PCI or PCI stress corrosion
4 cracking.
5 10 CFR 50 Appendix A GDC 10 requires that
6 specified acceptable fuel design limits, that's the
7 SAFDLs as we call it, are defined and it's demonstrated
8 that those SAFDLs are not exceeded during normal
9 operation and during AOOs.
10 So that's the regulatory framework that
11 requires that the licensees demonstrate that fuel
12 cladding does not fail during normal operation or during
13 AOOs. And that's reflected in guidance the staff has in
14 NUREG 0800 Standard Review Plan.
15 CHAIRMAN ARMIJO: Paul, I want to make
16 clear that it's the vendor's responsibility to identify
17 all of the failure mechanisms that are likely to cause
18 failure, and to address them either by analysis or
19 experiment or something. It's not the staff's job to
20 identify it for him.
21 MR. CLIFFORD: That's correct.
22 CHAIRMAN ARMIJO: So it's the vendor's
23 responsibility to assure that everything that could
24 cause failure, has a reasonable chance of failure, is
25 identified, analyzed and dispositioned.

1 MR. CLIFFORD: That's correct. That's a
2 good lead-in to the next slide. So as you mentioned, the
3 vendors need to identify all the degradation mechanisms
4 and failure modes. Here we have a list of eight failure
5 modes for fuel rod design. The first three have to do
6 with the fuel rod performance during an AOO, specifically
7 an AOO overpower type of vet.
8 You have your heat flux, which would be your
9 DNBR, fuel temperature, which would be fuel melt, and
10 cladding strain as a result of fuel thermal expansion.
11 The remaining five have to do with failure modes during
12 normal operation. Now in addition to these eight
13 SAFDLs, there would also be SAFDLs associated with fuel
14 bundle design.
15 So today in this presentation we'll be
16 focusing on these top three. And essentially, plants
17 rely upon the actual physical characteristics of the fuel
18 design in combination with operating procedures, LCOs,
19 especially initial thermal margin, automatic systems'
20 actuations, and manual response to preclude fuel failure
21 during normal operation and AOOs.
22 The Chapter 15 analysts are very good at
23 what they do. They hunt for the worst combination of
24 initial conditions, the worst combination of unfavorable
25 system response times, and they try to find that one

1 corner of operating space even though it may physically
2 not be able to co-exist.
3 In other words, they may have end of cycle
4 kinetics with beginning of cycle MTC. So they're always
5 searching for the limiting combination of initial
6 conditions to try to force the AOO to be the most
7 bounding.
8 And all I'm trying to show on this cartoon
9 on this figure is you've got the green box, which is
10 really where normal operation would be. They control
11 these plants in very tight windows. And then there's a
12 wider box around that as you increase power or maneuver
13 the plant. And that next barrier would be your LCOs.
14 That would be alarms in the control room.
15 And then the next box would be, that's based upon
16 very conservative models and 95/95 safety limits where
17 you would predict fuel rod to fail.
18 MEMBER BANERJEE: Is it always clear what
19 these limiting conditions are?
20 MR. CLIFFORD: The LCOs?
21 MEMBER BANERJEE: For the AOOs.
22 MR. CLIFFORD: Yes. For example, the
23 plants have alarms set up on all sorts of different
24 indications, and also those limiting conditions of
25 operation are defined in their tech specs. So they can

1 maneuver the plant, like actual power distribution, or
2 allow it a small window where they can maneuver the plant.
3 MEMBER BANERJEE: So they could go to the
4 limits of each of these, in theory?
5 MR. CLIFFORD: In theory they could.
6 MEMBER BANERJEE: And that combination is
7 the worst then?
8 MR. CLIFFORD: Well, they search to find
9 the worst. The worst may be the most --
10 MEMBER BANERJEE: So they actually do a
11 search?
12 MR. CLIFFORD: Oh, absolutely. The
13 analyst does that.
14 CHAIRMAN ARMIJO: Paul, just to go back to
15 Slide 6 is to keep in focus the contention of, at least
16 my contention and several other members, is that Item 8,
17 PCI/SCC, forget it's BWR, but that mechanism should be
18 moved up and put under the blue, because it is also
19 possible, and the concern is that it's, there's an
20 assumption being made by many people that if you meet the
21 DNB fuel melting and mechanical strain criteria you also
22 protect it from PCI. That's the issue.
23 MR. CLIFFORD: Right. And that's the
24 focus of today's presentation. I put it down here only
25 because I wanted to describe what our current practice

is.

CHAIRMAN ARMIJO: Sure.

MR. CLIFFORD: And we'll get to that in a second. Okay, so the question really is, would consideration of stress corrosion cracking during AOOs require further restrictions on the plant operations, its design, or on the tech specs?

Now just to get to what Sam was saying, the current regulatory practice is we only evaluate PCI as it relates to BWR plant maneuvering. When we review a fuel design, the last two I reviewed were the GE14E, which is for ESBWR, and the Westinghouse SVEA-96 Optima 2.

For those reviews we recognize that stress corrosion cracking has been seen. It's been experienced at BWRs because they have very strong control rods that can move individually, and when they move these control rods they get high local power changes which results in sustained cladding stress.

MEMBER BANERJEE: Can you just remind us what they put in the zirconium in these fuels, the Optima 2 and --

MR. CLIFFORD: Well, those were fuel designs --

MEMBER BANERJEE: They were not barrier fuels, correct? They didn't have --

1 MR. CLIFFORD: Yes, they were. They were
2 both barrier designs.
3 CHAIRMAN ARMIJO: Everybody in the BWR
4 industry --
5 MEMBER BANERJEE: They all had that.
6 CHAIRMAN ARMIJO: -- the liner. People
7 call it the liner now, but it's all zirconium with maybe
8 a pinch of iron or a pinch of tin or something like that.
9 The idea is to make --
10 MEMBER BANERJEE: Would they be tin or
11 iron, which one, can you remember?
12 CHAIRMAN ARMIJO: One vendor uses, two
13 vendors put iron, one vendor puts tin.
14 MR. CLIFFORD: Right now alloying the
15 zirconium barrier actually makes it perform worse from
16 a PCI perspective, but what it does is it helps you from
17 a secondary degradation. If you have a cladding
18 failure, say the debris fretting in a BWR that has barrier
19 cladding, that barrier cladding oxidizes very quickly
20 and it can result in long axial splits in the fuel rod.
21 So they try to alloy the barrier to make it less
22 susceptible to that oxidation which will make it harder,
23 which it makes it worse from a PCI perspective.
24 MEMBER BANERJEE: So it's a balancing act.
25 CHAIRMAN ARMIJO: But basically all of them

1 is to keep a soft layer on the inside of the cladding that
2 does not suffer the same extent of radiation hardening
3 as conventional zircalloy. So pure zirconium in the
4 highly irradiated state has the same yield strength as
5 zircalloy in the unirradiated state. So it stays soft
6 and ductile and relaxes instead of allowing that stress
7 to build up. That's how it works.
8 MEMBER BANERJEE: It doesn't form a barrier
9 against halogens and --
10 CHAIRMAN ARMIJO: No. No, it's totally,
11 you know, if you do a pressurized tube test with a
12 zirconium liner inside of it and you put iodine in there,
13 it'll fail by iodine stress corrosion cracking. It's
14 only in this particular mechanism of loading of the
15 expanding pellet during a power ramp where the stress
16 relaxation is fast enough in constraint and localized
17 enough that it relieves the stresses. So zirconium by
18 itself is not resistant to iodine stress corrosion
19 cracking.
20 CHAIRMAN ARMIJO: A lot of experiments --
21 MEMBER BANERJEE: Members of the
22 subcommittee need to have a little tutorial from you as
23 we go along.
24 MR. CLIFFORD: Okay. So I guess --
25 MEMBER BANERJEE: Not all, but one.

1 MR. CLIFFORD: So the takeaway here is the
2 current regulatory practices, were only focusing on
3 stress corrosion cracking as it revolves around BWR
4 control blade maneuvering.
5 MEMBER SHACK: But the language in the
6 standard review plan is more modest. So it says, you
7 know, none of these limits really protect you against
8 that. It doesn't claim to do it.
9 MR. CLIFFORD: Right. That is correct.
10 Okay, PCMI cladding strain. We mentioned
11 in a previous slide, during and AOO there are three
12 failure modes that we consider -- DNBR, fuel melt, and
13 cladding strain. As you move up in power you're going
14 to challenge all three of those. You're going to be
15 chewing away at available margins for all three of those.
16 And it depends on the type of accident which one is more
17 limiting. So we look at those three.
18 Now for each one of those three, it's up to
19 the fuel vendors to define what the SAFDLs are. So for
20 example, for cladding strain they would do mechanical
21 testing on irradiated cladding segments to determine
22 what the strain capability is of their particular
23 cladding alloy.
24 And that's a strong function of
25 manufacturing parameters such as composition as it

1 relates to corrosion resistance, heat treatment as it
2 relates to hydride orientation and distribution, neutron
3 fluence, and temperature.
4 They would also test -- well, I mean the
5 melting point of uranium is well known, but if they had
6 additives in their fuel they would also have to apply what
7 SAFDLs are with respect to fuel melt, and would also have
8 to do electronically heated testing on their assemblies
9 to come up with critical heat flux correlations, which
10 they would then use to develop a DNBR limit. So they
11 would define a SAFDL for DNB for fuel melt and for PCMI,
12 which is strain based.
13 Now this is just to remind me that we need
14 to go to the closed session, but we will be presenting,
15 or Patrick will be presenting results from over 200
16 Studsvik ramp tests on PWR fuel rods, and break it down
17 by various -- will look at it from different
18 sensitivities to determine how PCI is affected by burnup
19 or stress or anything else, time.
20 So with that, that was just the
21 introduction. Are there any questions on what was
22 presented before we have to go to the closed session?
23 CHAIRMAN ARMIJO: None. Let's go ahead.
24 Let's close the session, turn off the bridge link.
25 (Whereupon, the foregoing matter went off

1 the record at 1:19 p.m. and went back on the record at
2 2:58 p.m.)
3 CHAIRMAN ARMIJO: Okay, now for the good
4 part, right? Paul is going to tell us --
5 MR. CLIFFORD: Everything you wanted to
6 know about PWR --
7 CHAIRMAN ARMIJO: AOOs.
8 MR. CLIFFORD: -- AOOs. Okay, let me
9 begin. You know, operating experience has clearly shown
10 that PWRs are susceptible to PCI stress corrosion
11 cracking, and that's predominantly due to the way they
12 manipulate power in the reactor using very high worth
13 control blades.
14 When you move a control blade and expose an
15 area of previously controlled fuel, you're going to get
16 a sustained period of high power, high cladding stress.
17 PWRs have not been shown to be --
18 (Off the record comments)
19 CHAIRMAN ARMIJO: Sorry, Paul. I think
20 they're opening the bridge line. Just go ahead.
21 MR. CLIFFORD: Okay. Operating
22 experience has shown that PWRs are not susceptible.
23 There was limited number of stress corrosion cracking
24 failures due to missing pellet surface, but that issue
25 has been resolved through changes in manufacturing specs

1 and inspection procedures.

2 Okay, this slide breaks down the PWR fleet

3 based on NSSS design. There are six B&W 2-loop,

4 once-through steam generators, 12 combustion

5 engineering 2-loop, u-tube steam generators, five 2-loop

6 Westinghouse, 13 3-loop Westinghouse, and 29 4-loop

7 Westinghouse. These numbers already reflect the four

8 plants that shut down this year.

9 But the purpose of this slide is to show that

10 there is some differences in NSSS design but they're not

11 significant. The most significant difference between

12 the design and how the design responds to AOOs is really

13 the once-through steam generators for the B&W units.

14 They don't have the inventory as is the u-tube steam

15 generator, so there is certain design basis accents that

16 look significantly different.

17 CHAIRMAN ARMIJO: Different in more

18 demanding duty or --

19 MR. CLIFFORD: Well, a u-tube steam

20 generator has a lot more inventory on the secondary side,

21 for example. So a steam line break is a very significant

22 postulated accident for a u-tube because you have a

23 significant blow-down, because if you have a significant

24 return to power, whereas a once-through generator has a

25 lot less inventory so it blows down very quickly so it's

1 not a very severe accident.

2 And the same could be true about how it

3 responds to a feedwater line break. But those are all

4 postulated accidents which are not really the focus of

5 this.

6 CHAIRMAN ARMIJO: We're already in

7 trouble.

8 MR. CLIFFORD: Okay, PWR operating

9 conditions. For general reactivity control BWRs rely

10 upon control blade insertion to hold down excess

11 reactivity. That's how they get the two-year cycles.

12 PWRs, on the other hand, use boric acid dissolved in the

13 RCS to hold down excess reactivity.

14 So at the beginning of the cycle you may have

15 1,400 or 1,500 ppm of boric acid, and then it steps down

16 approximately three ppm per day throughout the reactor

17 life, or the cycle. It's not the reactor life but the

18 cycle life.

19 With respect to control rod insertion, as

20 we've previously mentioned BWRs will operate with

21 control blades inserted. These high worth rods may be

22 individually inserted and withdrawn. In contrast, PWRs

23 operate normally all-rods-out. Control rods are

24 assigned in banks and strictly restricted at full power

25 operation via the tech specs. Those are the power

1 dependent insertion limits.

2 And on the next slide, here's an example

3 from a PWR. If you look at this figure you see that only

4 Bank 5 at full power can be inserted about one-third of

5 the way into the core. Now Bank 5 would have four or five

6 different control rods going into four or five different

7 fuel assemblies.

8 CHAIRMAN ARMIJO: Paul, could you just sort

9 of walk me through? I'm looking at your chart there and

10 it's not obvious. I see the Group 5, and then the

11 fraction --

12 MR. CLIFFORD: Right, if you look, on the

13 y-axis is power. So if you go to the top it would be 100

14 percent power. Now you get out to the x-axis, this is

15 the different groups of banks. And you can see only Bank

16 5 is inserted. Here it's, how many inches is that?

17 Well, it's about one-third of the way inserted, so it's

18 about 30 inches.

19 CHAIRMAN ARMIJO: And how many fuel

20 assemblies are involved when that bank is --

21 MR. CLIFFORD: Four or five, depending on

22 whether it has a center rod. But there's only four fuel

23 assemblies, or five assemblies that would actually be

24 rodded, and they would only be rodded for one-third of

25 their active height.

1 CHAIRMAN ARMIJO: Okay.

2 MR. CLIFFORD: So because of this tech
3 spec, you can see that you don't have a situation where
4 you could be in where a movement of a control rod would
5 cause a significant change in power when you're at full
6 power conditions.

7 MEMBER BANERJEE: I don't see that. Can
8 you just do it again for me?

9 MEMBER REMPE: Maybe use the mouse to
10 point, because you were kind of pointing at the screen
11 and we couldn't see.

12 MR. CLIFFORD: Okay, so when you're at hot
13 full power you're limited by this tech spec to insertion
14 of only one bank that contains four or five rods, and
15 you're only allowed to be inserted into the core
16 one-third of the way.

17 And so the delta-rho per inch, the worth of
18 those rods is very small because you just don't have that
19 much bite because you're not inserted into the core. By
20 contrast, if you're at hot zero power you can have five,
21 four and three in.

22 So this event, the inadvertent bank CEA
23 withdrawal event is very limited from the zero power,
24 because you have a high worth rod. It starts to behave
25 more like a rod withdrawal from a BWR perspective. But

1 when you're 100 percent power you have very low worth
2 because of this restriction.
3 CHAIRMAN ARMIJO: So you could move them up
4 and down. It doesn't really matter much as far as --
5 MR. CLIFFORD: You can only insert them
6 one-third of the way into the core.
7 CHAIRMAN ARMIJO: But if you did a rapid
8 extraction at full power, let's assume --
9 MR. CLIFFORD: From one-third.
10 CHAIRMAN ARMIJO: From one-third. It
11 still wouldn't make much difference in power?
12 MR. CLIFFORD: No.
13 MEMBER BANERJEE: And it wouldn't make a
14 difference in local power?
15 MR. CLIFFORD: Not significantly, no.
16 MEMBER BANERJEE: Yes. I guess that's the
17 issue, right.
18 MR. CLIFFORD: Well, a control blade, you
19 can have a heavily controlled core in a BWR, and you move
20 one blade, the power in that one region of the core goes
21 up. The global power may not change, but the power in
22 those rods surrounding that moving blade goes up
23 significantly.
24 MR. BALLINGER: But there are maneuvering
25 restrictions.

1 MR. CLIFFORD: Right. And that's the
2 basis for those maneuvering restrictions.
3 MR. BALLINGER: And you're talking about
4 fine motion control rods or the old --
5 MR. CLIFFORD: Right. They've developed
6 controls to limit the worth of the rod so that they don't
7 get failure. But they have the capability of getting
8 failure because they have the worth of the blade. The
9 Ps --
10 CHAIRMAN ARMIJO: Doesn't have the
11 capability.
12 MR. CLIFFORD: Really don't have the
13 capability to have that worth, and they can't move
14 individually, and they're limited to only one-third of
15 the way insertion.
16 MEMBER BANERJEE: So the takeaway from this
17 is that they're not going to be large local power
18 distortions.
19 MR. CLIFFORD: Correct.
20 MR. BALLINGER: Under normal conditions.
21 MR. CLIFFORD: If the main culprit of
22 stress corrosion cracking in a BWR is maneuvering control
23 rods or control blades because of their high worth, you
24 just don't have that problem in a PWR because you don't
25 have individual high worth control rods that you can

1 move. You've got to move the bank and the bank is limited
2 to insertions such that is doesn't have high worth.
3 The next important operating
4 characteristic of a PWR are the reactor kinetics. In
5 addition to a negative FTC, fuel temperature
6 coefficients, PWRs operate with a negative MTC,
7 moderator temperature coefficient.
8 As I mentioned earlier, only at the very
9 beginning of a new cycle because of the high
10 concentrations of boric acid, may you have the
11 possibility of having a slightly positive MTC. But
12 that disappears almost immediately, and before you reach
13 full power you'll have a negative MTC. And that's a very
14 important characteristic because the reactor becomes
15 very stable.
16 If you increase power, the negative
17 feedback is going to want to decrease power. The same
18 is also true if you decrease power. It's going to want
19 to go back to where it was. Any increase in reactor power
20 without the same proportional increase in secondary
21 steam demand, all it's going to do is cause the water to
22 heat up, which is going to introduce negative reactivity
23 which is going to bring the reactor right back to where
24 it was.
25 So from a PWR the rule of thumb is the

1 reactor power always follows secondary steam demand. So
2 the secondary side drives reactor power.
3 CHAIRMAN ARMIJO: Before you get into the
4 steam generators, because I think this is a good time to
5 bring it up, in the BWR, one of the things that adds a
6 lot of reactivity in an AOO is a loss of feedwater heater
7 transient. It takes the whole core up.
8 MR. CLIFFORD: Right.
9 CHAIRMAN ARMIJO: If you had a boron
10 dilution event in a PWR that should also take the whole
11 core up, but the same feedback effects are in play that
12 you can't get a big power --
13 CHAIRMAN ARMIJO: Well, first of all, a
14 boron dilution of that would be very, very slow at full
15 power. Because you're relying on your letdown and
16 charging system which are very limited. So the rate of
17 boron dilution would be extremely slow.
18 CHAIRMAN ARMIJO: Okay.
19 MR. CLIFFORD: But yes --
20 CHAIRMAN ARMIJO: It's not the same rate at
21 which, you know, I'm saying if you lose a feedwater heater
22 and you've got six or seven or eight of them, I would think
23 that would be a slow transient too.
24 MR. CLIFFORD: For BWR?
25 CHAIRMAN ARMIJO: Yes.

1 MR. CLIFFORD: Well, for BWR, you know,
2 you're collapsing voids and doing things like that which
3 is causing quick increases in overall reactivity. PWRs
4 of course, they're decoupled so you don't have that.
5 CHAIRMAN ARMIJO: Okay, so the boron
6 dilution --
7 MR. CLIFFORD: The boron dilution
8 accidents, it's actually not even, most plants don't even
9 analyze it at full power because it's just not limiting
10 in the least amount. Boron dilution is really a lower
11 mode accident.
12 CHAIRMAN ARMIJO: Okay.
13 MEMBER SKILLMAN: Sam, boron's worth about
14 a 100 ppm of percent delta K over K, and say you're
15 offering, like Paul said, that you become critical of the
16 1,400 ppm. You're coming down three to four ppm per day,
17 and say you're 1,000 ppm. To get a large reactivity
18 increase you would really have to move a ton of boron.
19 But to do that you'd have to move an ocean of water. So
20 it is a very slow --
21 CHAIRMAN ARMIJO: So it's not even analyzed
22 as a significant AAO --
23 MEMBER SKILLMAN: Correct, because it's so
24 slow.
25 CHAIRMAN ARMIJO: -- so slow.

1 MEMBER SKILLMAN: It's just really slow.
2 On the other hand, yes, reactor power follows steam
3 demand, but a large increase in feedwater that is a
4 cooling event will drive you on moderator temperature
5 coefficient to a very high power level. So feedwater
6 issues on the P are potentially serious.
7 MR. CLIFFORD: Okay. And we're going to do
8 a survey of all the AOOs. But it's just important to take
9 away that PWRs are inherently stable and they always
10 resist power increases because of the negative FTC and
11 the negative MTC.
12 Here's a typical COLR MTC limit. And the
13 takeaway is essentially, for this particular cycle they
14 had a minus 0.2 at hot full power, and that would have
15 been maybe challenged on the first day of operation, and
16 after that it would be inherently more negative.
17 Really, for the PWRs it's the most negative MTC which is
18 somewhat limiting from an accident perspective.
19 Okay, I'm going to get into the reactor
20 protection system for PWR. If you have an overpower
21 event, which is of interest in this study, you have to
22 ask yourself, what are the protective features of the
23 PWR?
24 Well, all PWRs have safety-grade ex-core
25 detectors, and they provide a prompt indication of power

1 changes or rapid, a rapid indication of power changes.
2 And the PWRs all have what they call either a high flux
3 trip or a variable high power trip -- and we'll get into
4 the different designs -- which will provide protection
5 against any overpower event.
6 In addition, there are other driven safety
7 grade trip functions. Westinghouse and MHI have an
8 overpower delta T, CE and AREVA have a low DNBR. These
9 would be driven off of different instrumentation.
10 CHAIRMAN ARMIJO: That's in addition to --
11 MR. CLIFFORD: The high power trip.
12 CHAIRMAN ARMIJO: The neutron flux.
13 MR. CLIFFORD: Correct.
14 This slide shows you the tech spec allowable
15 values for the different designs. Basically they range
16 from 105 percent to roughly 110 percent power, the
17 different designs.
18 CHAIRMAN ARMIJO: These are much smaller
19 than the BWR transients, percent over power. I recall
20 seeing numbers, 115, 120 percent on BWR transients.
21 MR. CLIFFORD: In the GE plants, I'm not
22 familiar right now off the top of my head. I couldn't
23 tell you what the setpoints were.
24 CHAIRMAN ARMIJO: You know, these are
25 pretty, ten percent, five percent, pretty benign. But

1 the BWRs, as Bill would say, big honking.

2 MR. CLIFFORD: These setpoints are driven

3 by, it's not just the power. They have to provide the

4 trip in time to prevent the violation of the SAFDLs. So

5 if you have an event where you have an excess cooling

6 event, power's going up very rapidly.

7 A trip at 109 percent, you get your trip

8 signal when you're at 109 percent power, but the rods

9 don't start falling in until you're at 118, 120 percent

10 power. So you have to continually lower the trip

11 setpoint based on the speed of your coil decay time, your

12 instrument response time, uncertainties, et cetera.

13 CHAIRMAN ARMIJO: So that's the power at

14 which it trips, but what is the power achieved?

15 MR. CLIFFORD: It depends on the rate at

16 which it was climbing. There is, I would say, in

17 general, there's probably about a one second delta

18 between when you reach your trip setpoint and when your

19 rods start falling. You've got about a 0.4 second delay

20 time, and then there's a 0.6 second coil decay time. So

21 about one second after you reach your setpoint your rods

22 are actually free to fall.

23 CHAIRMAN ARMIJO: Okay, to the over-shoot

24 is, let's say, take a Westinghouse with a 110 percent trip

25 setpoint. What's the maximum power that actually would

occur with it tripping at that setpoint?

MR. CLIFFORD: It depends on the event. You know, the fastest rate of change in power is probably a large break steam line break, because you could have as much as a minus 4.0, 10 to the minus 4 delta-rho per degree Fahrenheit MTC which is driving power very fast. So maybe that event gets up to 118 or so.

CHAIRMAN ARMIJO: Okay, so it could over-shoot --

MR. CLIFFORD: No. As I mentioned earlier, the transient analyst will go off and he will find a combination of conditions to maximize that over-shoot. That's his job is to find the worst conditions to either totally avoid the trip, or if it's going to hit the trip to over-shoot the trip.

CHAIRMAN ARMIJO: Okay.

MR. CLIFFORD: Okay. Just what I said a second ago. Okay, so PCI stress corrosion cracking requires both stress and time. You need both components there. So is there an AOO overpower scenario which exhibits both of the necessary components? A prolonged power excursion that's going to give you sufficient stress for a sufficient period of time, can you find that scenario? And that's really the purpose of this investigation.

1 Before I leave this slide, the important
2 thing is if you over-shoot the reactor trip you're going
3 to get the trip function within a second. So if you
4 over-shoot your trip, you'll have sufficient stress but
5 you won't have sufficient time. So that in a sense
6 eliminates all of those events which rapidly over-shoot
7 the setpoint.
8 So you're in two families here. In order
9 to get enough stress for enough time, you really have to
10 avoid that trip. You're going to go off and find a
11 transient that goes up in power but for various reason
12 doesn't get that trip, and stays there for the 45 seconds
13 or 50 seconds that's necessary for crack propagation.
14 CHAIRMAN ARMIJO: Paul, I've got to bring
15 up something. John Stetkar couldn't attend but he did
16 send a question that I think fits right here. And he
17 starts out with, and so I'm going to read it to you because
18 it's his question.
19 And he says, "I'm not trying to raise
20 "academic concerns" like a 12 percent rated valve that
21 opens 75 percent of its full stroke, I'm mostly concerned
22 whether plants have automatic trip setpoints that are
23 above the power that would result from a single valve
24 operating fully, and whether prolonged operation at that
25 power level could cause PCI failures.

1 "For example, suppose that an automatic
2 reactor trip setpoint is 110 percent of rated power.
3 Suppose also that a steam generator atmospheric relief
4 valve or turbine bypass valve has a steam relief capacity
5 that is nine percent of full rated powers," full rated
6 steam flow --
7 MR. CLIFFORD: Right.
8 CHAIRMAN ARMIJO: "Would an automatic
9 reactor trip occur if one of those valves opens
10 spuriously? If so, what parameter would cause a trip?"
11 And then he says, "Does every PWR have an automatic
12 reactor trip that will occur if any one steam generator
13 atmospheric relief valve or any one turbine bypass valve
14 opens fully?"
15 There's a lot of stuff there so --
16 MR. CLIFFORD: So the answer is he's right.
17 It's about nine or ten percent. That's the relief base
18 capacity of one valve. So as we mentioned earlier,
19 reactor power follows secondary demand. So if you open
20 an atmospheric dump valve, now you have 109 percent
21 design steaming rate, reactor power's going to go to 109
22 percent.
23 The speed it gets there depends on cycle
24 burnup, which depends on the MTC. If you have an MTC
25 that's just slightly negative it's going to get there

very slowly. If at end of cycle you have a very negative MTC it's going to get there more rapidly. But it will go to 109 percent if a secondary steam event.

Now, so it's a race between whether you get a trip based on an overpower, say the setpoint's 110 so you stay just below it, but you'll also have an overpower delta T trip setpoint, and you'll also have potentially a low DNBR trip setpoint, but you'll also a low steam generator pressure trip setpoint.

So what the transient analyst will do is he'll find the MTC that gives you a slow enough increase in power so that you avoid your high power trip, and then you have to wait as long as it takes to get to the lowest steam generator pressure trip.

CHAIRMAN ARMIJO: Okay, and so we'll find out what that time is. And that's kind of the --

MR. CLIFFORD: And I have a plot of that.

CHAIRMAN ARMIJO: The last part of his question was, "If an automatic trip would not occur, when would the operators be expected to trip the reactor manually, if there is such a thing, and what is the basis for that response time?"

MR. CLIFFORD: Here's the exact power response time for the inadvertent opening of a steam generator atmospheric dump valve. I'll just jump ahead

1 to this. The transient analyst chose the initial
2 conditions which would avoid a quick reactor trip. But
3 as you can see, power goes right up to where the excess
4 steaming demand is, in this case 110 percent, and it stays
5 there for, this trips at approximately 200 seconds.
6 That's when they got the lowest steam generator pressure
7 trip. So this is the exact scenario that he was asking
8 about.
9 CHAIRMAN ARMIJO: Yes.
10 MR. CLIFFORD: So this is really, it's good
11 that he brought this up because this is really one of the
12 only postulated events that can get you to a high power
13 without getting a rapid trip. It's this or it's the
14 inadvertent opening of a turbine admission valve, you
15 know, opening it further. I mean there's a lot of
16 playing with the secondary side. But the only way to get
17 a prolonged power increase is on the balance of plant
18 side. You have to increase steam demand to reach a new
19 pseudo-steady state condition. CHAIRMAN
20 ARMIJO: So you reach the terminal power in about what,
21 four minutes, three minutes, something like that?
22 MR. CLIFFORD: Three minutes.
23 CHAIRMAN ARMIJO: Three minutes.
24 MR. CLIFFORD: And that would depend on
25 what MTC you assumed in your analysis. You could get

1 there a lot sooner if you assumed a very negative MTC.
2 CHAIRMAN ARMIJO: That's total core power.
3 Is every rod in the core moving up an incremental amount
4 at the same percentage increase or is it just the peak
5 rods?
6 (Simultaneous speaking)
7 CHAIRMAN ARMIJO: I mean fuel rod, fuel rod
8 power.
9 MR. CLIFFORD: Right.
10 CHAIRMAN ARMIJO: Let's say you have a
11 1,000 rods operating. Their peak node is 5 kilowatts a
12 foot, and there's 300 that are operating at 8 kilowatts
13 a foot. Where does the power come from? Do they all
14 move up a certain percentage?
15 MR. CLIFFORD: This would be a global
16 increase in power. There would be no significant
17 increase in local power. In other words, if you were
18 operating at 5 kilowatts a foot or you were operating at
19 10 kilowatts a foot, you would see roughly a nine percent
20 increase. So both get a nine percent increase.
21 CHAIRMAN ARMIJO: So from a PCI standpoint,
22 the rods that would be at greatest risk would be the ones
23 that were at peak power, because they would get a
24 percentage increase of maybe one kilowatt a foot.
25 MR. CLIFFORD: Yes and no. There's a lot

1 going on there. Because the fresh fuel is going to have
2 the highest power in the core, and the fresh fuel is going
3 to have an existing gap. So even though they go up nine
4 percent power they may not even close the gap. And
5 you'll see that from some calculations I have coming up.
6 CHAIRMAN ARMIJO: Okay.
7 MR. CLIFFORD: So it depends on a lot of
8 things, but initial gap size at the start of the transient
9 is very important from a stress perspective. So let's
10 go back to where we were. We'll get to that. We'll walk
11 to this logic.
12 MEMBER BANERJEE: So I just wanted to ask
13 you a question on that previous slide that you showed.
14 No, not that one. I mean what you went back to. Yes.
15 There you say that stress and time.
16 MR. CLIFFORD: Stress and time.
17 MEMBER BANERJEE: Is that what you find
18 when you look at your database where I had the impression
19 you were only talking about stress as a criteria, whereas
20 time has now appeared since that time. Now I don't know
21 if you can talk about it in open session, so --
22 MR. CLIFFORD: You're talking about a
23 gradual.
24 MEMBER BANERJEE: In broad terms talk about
25 it, yes.

1 MR. CLIFFORD: So PCI failure is driven by
2 both stress and time. You need to propagate that crack.
3 So the data shows that, yes, at a high enough stress you
4 can get failures. And I think the lowest time for
5 failure that was presented earlier was about 45 seconds,
6 at a very high stressed rod.
7 And there's a lot of data, separate effects
8 data too, on pressurized tubes that have iodine in them
9 where they've come up with failure time as a function of
10 stress. And it's a very clear relationship. The higher
11 the stress, the less time. The lower the stress, the
12 more time.
13 MEMBER BANERJEE: So in talking about a
14 threshold, you really don't have a threshold. You have
15 a stress and time criteria. Is that it?
16 MR. CLIFFORD: I think that's where we
17 would like to go. Well, hopefully with SCIP-3 we'll have
18 enough information from an additional ramp test and from
19 the separate effects testing where we can have a
20 stress/time. You can get them both. Right now we don't
21 have that. We're just assuming stress. A threshold, a
22 low threshold. But we recognize there is a time
23 component, and if you get a reactor trip within two or
24 three seconds --
25 CHAIRMAN ARMIJO: Don't worry about it.

1 MR. CLIFFORD: -- don't worry about it.
2 You need time, so it really, it gets back to that.
3 There's two types of accidents we're going to
4 investigate. One, where you rapidly shoot through that
5 trip and you get your trip very quickly, and the other
6 one where you hang up just below a trip for enough time
7 to get that crack to propagate.
8 CHAIRMAN ARMIJO: Initiate. You know,
9 these cracks start with totally crack-free ID surfaces.
10 You don't need a crack starter for PCI to occur. So
11 you've got to initiate it and propagate it through-wall.
12 And it can happen in a minute.
13 MR. CLIFFORD: Right. It can happen in a
14 minute.
15 MR. BALLINGER: The stress at time really
16 translates into a strain rate at this crack tip that has
17 to be maintained for a certain amount of, to get
18 through-wall. So the stress and time are just
19 surrogates for a crack tip strain rate basically, so the
20 iodine or whatever the embrittling agent has to do is,
21 has to get access to the surface by fracturing, by getting
22 access. So it's really the strain rate.
23 So that's going to be a function of burnup
24 because the strength of the cladding's going to be a
25 function of burnup. So there's a lot going on here. You

1 can see why these, there's uncontrolled variables all
2 over the place here.
3 MR. CLIFFORD: Right.
4 MEMBER SHACK: But just in a macroscopic
5 sense, the FALCON code does a damage index where they do
6 an integral over the time and stress. The EDF and NRC
7 have chosen to do a threshold stress thing, which you can
8 do if you make it conservative enough. Because he's not
9 really interested in things that take an hour here. He
10 knows he's going to terminate these things.
11 MR. BALLINGER: Well, if the stress is high
12 enough you simply can't relax it fast enough before you
13 get a through-wall crack.
14 MR. CLIFFORD: So here we have a list of all
15 the AOOs which are evaluated, and I have two columns.
16 One whether it results in a local power increase and one
17 whether it results in a global power increase.
18 The first five events are all involved in
19 increase in secondary heat removal. And as I mentioned,
20 these plants operate with a negative MTC, so as you draw
21 more heat away and you start cooling down RCS moderator
22 temperature you're going to get an increase in power. So
23 those are some of the events we'll be looking at in
24 further detail.
25 The next events involve a decrease to

1 secondary heat removal, so these involve generally a
2 turbine trip or loss of condenser vacuum where you remove
3 heat, you stop removing heat from the primary side.
4 Everything heats up, and with the negative MTC you're
5 introducing reactivity instead of positive reactivity.
6 The next event that involves a power
7 increase is the inadvertent bank CEA withdrawal, and
8 we'll talk about that in more detail. As we mentioned,
9 boron dilution is really a lower mode event.
10 Control rod drop results in a small, very
11 localized change in peaking factors, but no global
12 increase in power as you can imagine dropping a rod. The
13 PPCS is a pressurizer, pressure control system
14 malfunction, and that involves either inadvertently
15 turning on your pressurizer heater or sprays. That's
16 really a pressure malfunction event. No changes in
17 power. Same is true for the CVCS malfunction. That's
18 a change in letdown and charging.
19 Inadvertent startup of inadvertent reactor
20 coolant pump is a lower mode event. These are operating
21 with all their pumps at full power. And ASGT would be
22 a closure of a main steam isolation valve on one of the
23 loops, which can involve changes in power locally in the
24 core but no global increase. So from looking at
25 the survey, I went through that fast but we're going to

1 focus on two types of events. The increase in secondary
2 heat removal events and the inadvertent bank withdrawal
3 event, because those are the two that are going to
4 potentially challenge PCI.
5 CHAIRMAN ARMIJO: And the whole core. For
6 example, if you had a control rod drop in the BWR, it's
7 easy to understand it drops. In the P we're talking
8 ejection?
9 MR. CLIFFORD: No, it's a drop. It's an
10 individual bank, say there's a malfunction of the control
11 rod drive system and it just loses magnetic field and just
12 falls. It falls into the reactor so it adds negative
13 reactivity into the core. Brings core power down, but
14 you can have a local change in peaking factors which you
15 have to look at.
16 CHAIRMAN ARMIJO: Okay, but that's very
17 local to the five assemblies that are affected. Okay.
18 MR. CLIFFORD: Well, it's not a bank. It's
19 an individual rod that falls.
20 CHAIRMAN ARMIJO: Oh, that's an
21 individual, one assembly.
22 MR. CLIFFORD: So we'll start with the bank
23 CEA withdrawal. Now the most limiting scenario for the
24 bank CEA withdrawal is one that tries to maximize the
25 over-shoot. And that's done by selecting the maximum

1 worth, the least negative MTC, least negative FTC, and
2 the minimum delayed neutron fraction.
3 So that event would result in a very rapid
4 trip. Now it is possible to identify a bank withdrawal
5 scenario, say a very low worth bank or partially
6 inserted, like just a few inches, which could cause a
7 relative global increase in power without getting that
8 trip.
9 However, you still need to recognize that
10 with no increase in secondary steam demand, eventually
11 reactor power will return to where it was because it's
12 driven by turbine power. And I just note here that it's
13 about an eight- to ten-second loop transition time.
14 So if we look at, and we'll start here.
15 Here's a trace of reactor power for the bank CEA
16 withdrawal from hot full power. This is the maximum rate
17 of increase in the very conservative modeling of this.
18 As you can see, power is not moving very
19 fast, and this is because the tech specs, remember, limit
20 rod insertion. You just don't have a lot of worth. And
21 so even when you maximize all your parameters you don't
22 get a very aggressive accident scenario.
23 And as I mentioned, this one gets a trip at,
24 looks to be about 14 seconds. So if you would try to run
25 this again and avoid that trip it would be even more

1 benign. It may be extended, but it would be even more
2 benign.
3 So kind of looking at this event, you say,
4 okay, well, how susceptible is this particular event to
5 stress corrosion cracking on a core-wide basis? And I
6 would say you'd have to conclude it's very low. First
7 of all, plants operate all-rods-out all the time. So if
8 you don't have control rods in the core you can't have
9 a control rod withdrawal accident. So the probability
10 is very low.
11 As we mentioned, the COLR limits insertion
12 so that your delta-rho per inch is very low. As a result,
13 your power excursion and your local peaking factors are
14 very low. You have a negative MTC which is going to
15 resist your power excursion. You have a trip function.
16 Your ex-core detectors, which are going to
17 trip you out at somewhere between 105 and 110 percent
18 power, so that's going to limit the duration and the
19 overall power increase you can get. And there's also
20 other safety grade trips that are available.
21 And we can't forget that if we try to come
22 up with a very benign event that just slowly creeps up,
23 there's obviously reasonable operator actions. There's
24 enough instrumentation in the control room to warn the
25 operator this is going on --

1 CHAIRMAN ARMIJO: So the 18 seconds does it
2 for me, Paul. I'm sold with the 18 seconds.
3 MR. CLIFFORD: Okay. So now we're going to
4 go into the next category of events. Excess steaming.
5 I call it excess steaming, but as was mentioned earlier,
6 it could be a decrease in feedwater temperature would
7 also cause an increase in heat removal.
8 So these are the type of events that I think
9 are the ones that we need to focus on because this is the
10 only type of event that's going to cause a pseudo-steady
11 state to occur. As we looked at the survey, there were
12 five AOOs. They're all evaluated. Generally, there's
13 two that are bounding, and that's the inadvertent opening
14 of the steam generator atmospheric dump valve and the
15 steam bypass control system malfunction.
16 And depending on the rate of excess heat
17 removal and the MTC there's a wide possibility of
18 scenarios that will occur. So you can find the right
19 combination that will give you a prolonged overpower
20 scenario, the magnitude of which we'll talk about.
21 So we already talked about this. This is
22 the 11 percent, or somewhere between nine and 11 percent,
23 the one atmospheric dump valve that opens up. Here's
24 another event that's much more severe, steam bypass
25 control system. These are the steam bypass valves.

1 Plants generally have somewhere between 40 and 90 percent
2 steam bypass capability. The newer plants like Palo
3 Verde, I think, had 88 percent. That's the most I've
4 seen.
5 So if you inadvertently open all these steam
6 bypass valves, it's almost like it's a small steam line
7 break. So you can see power increases rapidly here.
8 But the faster the power increases, the faster you get
9 a trip. So here, even though power may get to, I think
10 it's about 116 percent or so, I mean the event's over
11 within ten seconds.
12 So this isn't really the event you have to
13 worry about for PCI because it's too fast. So you really
14 have to go back to an event where you get a very benign
15 increase in power that hangs below those trip setpoints.
16 CHAIRMAN ARMIJO: Yes, it would be nice to
17 plott the IOSGADV and the SBCS on the same time scale,
18 and you can really see how different they are.
19 MR. CLIFFORD: Oh yes, you wouldn't even
20 see it on the --
21 (Simultaneous speaking)
22 CHAIRMAN ARMIJO: You wouldn't even see it.
23 It's a blip.
24 MR. CLIFFORD: Okay, so I think when you
25 start looking at PCI susceptibility for these type of

1 events, it really goes back to some of the same issues.
2 You have a trip function, and even though that trip
3 function wasn't designed for PCI, that trip function was
4 designed to protect DNB, it was designed to protect fuel
5 melt and cladding strain, it still appears that that trip
6 function is low enough to prevent stress corrosion
7 cracking in PWR.
8 And every NSSS is slightly different, and
9 you'll have slight differences, but they all have almost
10 identical trip functions, somewhere between 105 and 110
11 percent. And we saw that.
12 And in addition, can you really achieve a
13 long duration transient? You know, I mean is it
14 reasonable to assume that the operators are going to do
15 nothing if they see power go up and all these alarms are
16 going off?
17 And second of all, how long can you, I can
18 postulate with my codes, my crude codes, that you can
19 achieve it. But my crude codes don't model the turbine.
20 They basically have steam going off into nowhere land,
21 and an infinite amount of feedwater coming back in. It
22 doesn't model the entire loop on the secondary side.
23 So in real life, what's going to happen?
24 And you can't dump steam into your condenser without
25 losing condenser vacuum. So you would trip the plant on

1 condenser vacuum. You can't open an atmospheric dump
2 valve for a long period of time without causing the
3 reactor operators to take immediate action.
4 CHAIRMAN ARMIJO: Paul, when you say long
5 period of time, are we still talking that 100, 200
6 seconds?
7 MR. CLIFFORD: Yes, 200 seconds. You
8 can't dump that much steam into your condenser for 200
9 seconds without something causing you to trip. But
10 you're going to get a trip anyway, so I mean --
11 CHAIRMAN ARMIJO: So, you know, you're
12 saying there's margin there that you don't analyze for
13 in the way the overall system --
14 (Simultaneous speaking)
15 MR. CLIFFORD: Right.
16 CHAIRMAN ARMIJO: -- response.
17 MR. CLIFFORD: We generally don't take
18 credit for any operator action, and we don't take credit
19 for what we call equipment protection trips. Those are
20 reactor trips which are designed to save the equipment,
21 to save your turbine, to save your condenser, to save all
22 that other stuff.
23 We only take credit for safety grade trips.
24 So when we do take credit for safety grade trips then
25 these events are terminated too early for PCI to be a

1 concern.

2 CHAIRMAN ARMIJO: So if that is the

3 limiting transients, the worst one you can come up with,

4 that's the one you analyzed further to see what the

5 stresses are and --

6 MR. CLIFFORD: I'll get to that.

7 CHAIRMAN ARMIJO: Okay. But is that the

8 worst one that you've found?

9 MR. CLIFFORD: That's the only one that I

10 can think of that would give you an increase in power for

11 a prolonged period of time. But even that, my codes

12 wouldn't predict whether they would trip on something

13 else. In other words, you can't open an atmospheric dump

14 valve and expect this balance of plant to just operate

15 as if everything's going fine. You can't

16 postulate an inadvertent opening of a turbine admission

17 valve without then changing your stator bar, voltage, and

18 sync into the grid and doing all this other stuff. You

19 just can't increase turbine power, main turbine power,

20 without doing something.

21 CHAIRMAN ARMIJO: Well, if somebody wanted

22 to go through the steps they could find where other things

23 would trip and terminate the event.

24 MR. CLIFFORD: Correct. So it really is.

25 It really all hinges down on the reactor protection

system. Is that trip low enough? I mean that's the question we're going to answer next. Is that trip function low enough so that if you somehow found some event that stayed right below it for a long period of time, would you be in an area of stress concentration to give you PCI? So those are the questions we're going to answer.

Okay, so are there any questions on how a PWR operates or the reactor protection system or the survey of AOOs?

MEMBER SKILLMAN: Yes, I do have one question, Paul. And that is, say you're out on a 24-month fuel cycle, you're out 18 months, and you begin to trim on your axial power shaping rods. So you've had 18 months of soak, your boron concentration's down to 300, 350, something like that. You've got six months to go, and you decide just to even up on your APSRs or whatever the shim rods are that are with the other plant designs.

That fuel has been preconditioned for months, and now you move axial power shaping rods maybe ten inches, 20 inches, quite a bit. And I'm wondering if that isn't a PCI opportunity in the PWR.

MR. CLIFFORD: I guess, I've worked at two different PWRs. We never operated with axial shaping

1 rods in the core, so I'm not sure where you're going.
2 Where's the experience of operating with those rods in?
3 I mean they're only allowed by tech specs to be in a couple
4 inches, you know, 30 inches.
5 MEMBER SKILLMAN: APSRs are generally out
6 quite a bit, at least that was my experience at TMI. We
7 operated, now and when we got an axial offset we trimmed
8 with APSRs.
9 MR. CLIFFORD: I guess I'm not, I can't
10 really answer because I'm not that familiar with that
11 operation.
12 MEMBER SKILLMAN: And I would think all the
13 other Ps had at least some ingredient of that same
14 experience where you're actually trimming your axial
15 power shape.
16 MR. CLIFFORD: We didn't do that at the CE
17 plants.
18 MEMBER SKILLMAN: Maybe it's unique to the
19 B&W plant.
20 MR. CLIFFORD: Maybe it's unique to the
21 B&Ws.
22 MEMBER SKILLMAN: But I know that we did
23 that. I did that for years.
24 MR. CLIFFORD: Were those grey rods? Were
25 they like inconel rods?

1 MEMBER SKILLMAN: They were 25 percent
2 absorber and 75 percent non-absorber.
3 MR. CLIFFORD: Okay, so they were full
4 strength part length.
5 MEMBER SKILLMAN: They were full length
6 part poison. And their official name in tech specs is
7 axial power shaping rods, APSRs.
8 MR. CLIFFORD: Okay. Well, I mean, I guess
9 the only thing I would say is it sounds like they have
10 a lot of experience with that and they've never --
11 CHAIRMAN ARMIJO: Yes, but that's normal
12 operating --
13 MEMBER SKILLMAN: That's normal operation.
14 But the question was, is there an opportunity in the P
15 to have PCI/SCC? And Paul's saying, hey, we've looked
16 at all of these events. It's these feedwater events.
17 And I'm suggesting perhaps, at least in one
18 design, there is another actual operating occurrence
19 where you adjust on your axial power shaping rods where
20 they can be PCMI or PCI/SCC. And that's what I'm asking.
21 MR. CLIFFORD: Right. Well, I mean as far
22 as I know they have never failed a rod by moving these,
23 so from a maneuvering perspective they must have
24 developed very good guidance on the speed at which they
25 move these and when they move these so that they don't

1 fall into the --

2 MEMBER SKILLMAN: These are not for

3 maneuvering. These are for shaping.

4 MR. CLIFFORD: Right, that's what I meant,

5 power shaping for the flux. Yes, the CE fleet has, we

6 have part length rods, part strength rods to use for axial

7 control, axial power control, but they were never in for

8 any period of time.

9 MEMBER SKILLMAN: That's the difference

10 between the two designs then.

11 MR. CLIFFORD: Yes. I'll have to look into

12 that. So you're wondering if there could be an

13 inadvertent withdrawal of those, you know, beyond your

14 existing operating guidance, and that could cause the

15 stress corrosion cracking?

16 MEMBER SKILLMAN: Well, I guess what was

17 churning in my mind is, have we ever seen a PCI event that

18 occurred as a consequence of changing axial power shaping

19 rod position? And I know that the B&W plants went

20 through this when we began to see axial offset anomaly,

21 AOA. We began to move those rods.

22 MR. CLIFFORD: I had no knowledge of any

23 PWRs failing during maneuvering of these rods.

24 CHAIRMAN ARMIJO: I used to track PCI

25 failures in all systems many years ago. I kept feeling

1 very sorry for myself because a BWR had so many PCI
2 failures and the PWRs didn't.
3 And the very few PWR fuel failures that were
4 reported as being PCI were very rare. And at that time
5 I can tell you nobody was paying that much attention to
6 the issue of pellet chips and this missing pellet surface
7 thing. It wasn't even recognized that it could happen.
8 So my suspicion is that the very few PWR
9 "PCI" failures that were reported were related to
10 defective pellets. The system is very mild in normal
11 operating space, and the question is, is it just as mild
12 in AOO space, and that's where we're at now.
13 MR. CLIFFORD: Right. But to get back to
14 your concern, I have no knowledge that they failed any
15 rods while moving those, fuel rods, while moving those
16 things though.
17 MEMBER SKILLMAN: Okay, thank you.
18 MR. CLIFFORD: Okay, where were we? Okay,
19 now we're going to get into the FRAPCON calculations.
20 Okay, there were two different sets of calculations that
21 I performed.
22 The first one was intended to answer the
23 question, are the current setpoints sufficient to
24 prevent fuel failure during core-wide AOOs? In other
25 words, if you assume that you need a prolonged transient,

1 the only way to get a prolonged transient is to avoid that
2 trip, and essentially you have a ceiling on power.
3 And if you run cases at different burnup
4 time steps for different rod power histories with the
5 assumption that you jumped in power right up to that trip
6 setpoint, what's the cladding stress? And then if you
7 compare that against the researched, the empirical value
8 that we've developed that Patrick talked about, how does
9 it compare?
10 And in the second calculations, what I did
11 was, if I just ignore the trip function and then I just
12 iterate on power, changes in power, until I get to those
13 thresholds, what power would that be?
14 Okay, so the assumptions for this analysis
15 was that a rapid power excursion which over-shoots the
16 setpoint as we talked about would yield high stress,
17 however, wouldn't have the time duration necessary for
18 the crack to go through-wall. As a result, the prolonged
19 power excursion must remain below this trip setpoint, so
20 an instantaneous ramp to 112 percent of the initial rod
21 power was assumed.
22 CHAIRMAN ARMIJO: That includes the
23 over-shoot?
24 MR. CLIFFORD: No, it doesn't include the
25 over-shoot. I chose 112 percent because there may be

1 some temperature shadowing of the ex-cores for a cooldown
2 event which you've got to account for. So 109 percent
3 trip would be roughly a 111 percent trip because the
4 colder water is shielding the ex-cores. So I just threw
5 in a couple of percent to cover for that.
6 Okay, I used some prior rod power histories
7 that we had from one of the EPUs that we evaluated last
8 year. For that particular licensee, we identified the
9 highest second-cycle fuel rod and the highest
10 third-cycle fuel rod.
11 And we also did an analysis on the bounding,
12 what they call the bounding radial fall-off curve, which
13 all fuel rod power histories are compared to the fall-off
14 curve to show that they're below it. So it's the other
15 threshold.
16 CHAIRMAN ARMIJO: Paul, for those two cases
17 of limiting second cycle and limiting third cycle, what
18 is the peak power in kilowatts per foot?
19 MR. CLIFFORD: You'll see them on the
20 table.
21 CHAIRMAN ARMIJO: Yes. I mean humor me and
22 tell me the numbers. I just always have a problem. In
23 Bs I can tell you what that would be, but Ps I never know.
24 MR. CLIFFORD: Here, if you look at the
25 third column, right here, this would be the initial power

1 history.

2 CHAIRMAN ARMIJO: So roughly nine

3 kilowatts max.

4 MR. CLIFFORD: For these different rod

5 power histories.

6 CHAIRMAN ARMIJO: And they go down in the

7 third cycle, which makes sense. And then the radial

8 fall-off I don't understand.

9 MR. CLIFFORD: These aren't the same rods.

10 So there's a power history. When this licensee did their

11 EPU fuel management study they basically go out five

12 cycles. They look at different fuel management

13 strategies, and they do all these depletions to try to

14 get a feel for what the fuel costs are going to be for

15 the EPU.

16 So more, it goes into their business

17 decision. So I asked them to survey all of their fuel

18 management patterns that they had generated, and to find

19 the rod which in its second cycle had the highest power,

20 and then to find me another rod, which in the third cycle

21 had the highest power. Because when I was evaluating

22 this EPU, those are important limiting rods from a rod

23 internal pressure fission gas release perspective. So

24 that's the rod, you can kind of glean what the rod power

25 histories are.

1 Okay, so those are the rod power histories
2 we have. And as Patrick talked about earlier we have two
3 failure thresholds. Based upon the empirical database,
4 there's a 200 megapascal hoop stress as a lower bound and
5 a 250 megapascal as more of a best estimate.
6 MEMBER REMPE: Can you tell us again what
7 the bounding radial fall-off curve is, in more detail?
8 MR. CLIFFORD: Okay, they don't
9 necessarily perform fuel rod thorough mechanical design
10 calculations for every cycle. That sometimes they would
11 do an analysis and they come up with an envelope, rod
12 power versus burnup.
13 And as long as all the rods for the upcoming
14 cycle fell below that then the bounding calculations
15 would remain valid. So it's a reload checklist
16 parameter. So it wouldn't be one rod, it would be the
17 composite worst rod at every burnup point when we're
18 looking at it.
19 CHAIRMAN ARMIJO: Is that a real rod or is
20 that some sort of a fictitious power?
21 MR. CLIFFORD: It's a totally fictitious
22 power.
23 CHAIRMAN ARMIJO: So if we went back to the
24 limiting second cycle, the highest power initial LHGR is
25 8 kilowatts a foot on your table.

1 MR. CLIFFORD: That's not the highest power
2 rod in the core. There would be a first cycle rod that
3 might have a higher power early in life, but because it
4 depleted quicker in its first cycle it's second cycle was
5 lower in power.
6 So to basically get to the highest rod that
7 had the highest power in its second cycle it was probably
8 in a more benign location in its first cycle. And the
9 third cycle one is even --
10 CHAIRMAN ARMIJO: A first cycle rod lets a
11 mid-cycle, would have plenty of power.
12 MR. CLIFFORD: Correct.
13 CHAIRMAN ARMIJO: And it would be
14 susceptible to PCI. I mean it'd have a sufficient
15 burnup. Is that the radial fall-off thing or is --
16 MR. CLIFFORD: Yes, we looked at various
17 powers at the radial fall-off curve so that would bound
18 that.
19 CHAIRMAN ARMIJO: Okay, so we're talking
20 about something, let's say in the worst case, maybe
21 somewhat fictitious, about nine kilowatts a foot as an
22 initial, nine, nine and a half?
23 MR. CLIFFORD: Right. Okay, we're on Page
24 33 now, the table?
25 CHAIRMAN ARMIJO: Yes, that's what I'm

trying --

MR. CLIFFORD: Right. The radial fall-off curve, if you look at this burnup, is 31 gigawatt-days. That has a 9.55 kilowatts a foot and that was the worst.

CHAIRMAN ARMIJO: That'd be the worst rod, but there'd be more than one. There'd be --

MR. CLIFFORD: Oh absolutely.

CHAIRMAN ARMIJO: Okay, so that's that one. Okay.

MR. CLIFFORD: Okay, so we're on this slide so we might as well try walking through it. Okay, so there are three power histories, so three different rods, although the last one is a composite worst case. And what I did was simulate a 12 percent increase in local power at different burnup time points.

And what's interesting is, if you look at the first results here you have a hoop stress, which is compressive in nature, which would indicate that that gap would be open and that would make sense at the beginning of life. So you have compressive load as a result of the RCS pressure. And when you perform an instantaneous ramp at 12 percent nothing really happens because you really haven't closed the gap. And that's what you would expect.

As you go up in burnup, here you're at 20

1 gigawatt-days, your pressure is still the same. So it's
2 really, you need to get up to about, for this particular
3 rod you needed to get up to almost 40 gigawatt-days before
4 you saw any increase.
5 So here you started with a gap, but you've
6 closed that gap just barely. And as you can see, towards
7 the end of its life at 48 gigawatt-days you started with
8 a tensile load as a result of PCMI just as a result of
9 normal operation, and you saw a pretty dramatic jump in
10 cladding hoop stress as a result of that 12 percent jump.
11 So the maximum for this particular would be 120
12 megapascals, which is well below even the 200 megapascal
13 lower bound threshold. Now we go to the next,
14 this was a limiting third cycle rod. I started with
15 higher burnups because I wanted to make sure that we
16 started with the gap closed, because if it was open it
17 didn't really make too much of a difference.
18 And as you can see, the maximum change
19 occurred, and the maximum hoop stress is at the high
20 burnup, 58 gigawatt-days, and that's 160.5 megapascals.
21 If we go down to the radial fall-off curve, which is a
22 fictitious fuel design but has high power at high burnup,
23 you can see that the calculated hoop stress in all cases
24 was below 200 megapascals, which is the lower bound and
25 certainly below 250 megapascals, which was the best

1 estimate failure threshold.

2 So this investigation shows that the

3 predicted cladding hoop stress remains below these

4 failure thresholds for this prolonged overpower

5 scenario. So the conclusion is that the calculations

6 suggest that the existing high power trip setpoints are

7 sufficient to prevent PCI since they limit power to below

8 roughly 110 percent power.

9 CHAIRMAN ARMIJO: Well, if you took the

10 stress-strain curve for the irradiated zirconium

11 cladding alloy, say you had one, and you plotted your 230

12 megapascal or whatever your threshold value was, and you

13 came down parallel to the youngest modules to get down

14 to sort of, is it purely elastic loads or does it have

15 a plastic -- yes, I'm looking for a threshold based on

16 strain, and you have a stress-strain curve and you've got

17 a stress so you ought to be able to tell me what the strain

18 is. So what would be the strain that's commensurate with

19 that stress threshold?

20 MR. CLIFFORD: Right. And I have to look

21 that up, but I'm not sure what we would glean from that.

22 CHAIRMAN ARMIJO: Yes. Well, no, but

23 here's the argument. Let's say it's a strain. If I have

24 data that says fuel rods fail at low strains if you have

25 a PCI environment and sufficient power increase, and that

1 strain threshold for failure is 0.1 percent instead of
2 1 percent, okay, that would tell you you have to reach
3 a certain stress.
4 But if you start with a stress, let's say
5 you've picked a stress as the threshold but you could have
6 picked a strain and calculate the strain. And I'm just
7 trying to get an idea of what is the strain that is
8 consistent with that stress value you picked?
9 MR. CLIFFORD: Right, I have to look into
10 that.
11 MEMBER SHACK: FRAPCON presumably spits
12 that out.
13 MR. CLIFFORD: Oh yes. No --
14 CHAIRMAN ARMIJO: It's there. It's
15 somewhere. You know, if it was 0.01 percent or all
16 essentially elastic, I'd say you haven't got any strain
17 so you don't have a problem. But if it's substantially
18 more than let's say 0.1 percent, I think you're not out
19 of the woods yet.
20 MR. CLIFFORD: Obviously, like as you
21 mentioned, it's there in the outputs. I just didn't plot
22 it.
23 CHAIRMAN ARMIJO: Yes.
24 MR. CLIFFORD: I would imagine it's very
25 low.

1 CHAIRMAN ARMIJO: I don't have the stress
2 rate curve in my mind of what it would be. Bill's
3 probably going to find it for us.
4 MEMBER SHACK: Well, I'm just sort of
5 looking at what they calculated for the, you know, for
6 the sort of 250 was up there around 0.4 percent plastic
7 strain, so it's well below that. Just how far below
8 that, you know, you really have to start getting a little
9 bit more precise than anything we can sort of whip off
10 the top of our heads here, I think.
11 MR. CLIFFORD: Right.
12 CHAIRMAN ARMIJO: But, you know, again we
13 have data, very well characterized rods. But you can't
14 measure the stress in a ramp test. All you can do is
15 power, delta power, burnup, and finally the strain, which
16 post test strain.
17 If FRAPCON could, do the same FRAPCON
18 calculation you have now only your inputs are different,
19 you know, you should be able to come up with what is the
20 calculated strain and see if it comes close to the
21 measured failure strains from the BWR database, which is
22 very well characterized and it's not confounded by all
23 of the variables that Patrick raised.
24 And so that, to me that's just waiting there
25 to be used. And I think you could put it to bed if you

1 don't --

2 MR. CLIFFORD: I mean I think the

3 calculated experience is going to be very low, because

4 even in the over-shoot scenarios where you go up to 120

5 percent power, you don't come anywhere close to the one

6 percent. I know the one percent you're saying is not the

7 right number, but you don't get close to those at all.

8 CHAIRMAN ARMIJO: You get over 0.1.

9 MR. CLIFFORD: You may.

10 CHAIRMAN ARMIJO: And if you look at the

11 white paper on the BWR rods it's 0.1, 0.15 percent, 0.2,

12 and that's where they're failing. But again, I'm saying

13 the data, you know, I'm not just tooting my own horn of

14 the test rods that we did, but the Kraftwerk Union did

15 totally independent of the GE tests, did the same tests,

16 and it was remarkable how close their results were. So

17 I think a strain criteria would work, or at least could

18 check whether FRAPCON is really predicting what will

19 happen.

20 And in the BWR, I think you're going to get

21 more strain because you don't have that external pressure

22 working against you, and we don't, the gaps closed for

23 these rods, I'm pretty sure. So, you know, FRAPCON would

24 have, you have a lot of variables that you have to put

25 as input, but everything is known about those rods, so

1 I would be very --

2 MR. CLIFFORD: Okay, I mean --

3 CHAIRMAN ARMIJO: Yes, think about it.

4 Think about it. I think it could help put this to bed.

5 Because, you know, I'll tell you, just looking at your

6 table on Page 33, all of those rods that the ramp, LHGR

7 was six, six and a half, I can tell you, if these were

8 BWR rods nobody would even blink because it's way below

9 the eight kilowatt a foot threshold. Even if it's

10 diffuse, nothing has ever failed below seven.

11 And so, you know, that tells you that's not

12 a problem area. Around seven and eight, it's in this

13 threshold stuff. The stuff around ten, if those were

14 real fuel rods that's plenty of LHGR to fail BWR rods.

15 But, you know, if those are just calculated numbers that

16 don't represent a real fuel rod, I don't know what to do

17 with those.

18 MR. CLIFFORD: Well, but it's not just the

19 linear heat generation rates. You know, what we're

20 trying to take away here is that you've got a trip

21 function that's going to limit it to a very small delta

22 increase. And the question is, during that delta

23 increase is there enough stress or strain to cause a

24 failure?

25 And, you know, like I said, the takeaway is

1 that you have this trip function that's set just five,
2 ten percent above where you're operating. So you just
3 don't have the delta.
4 CHAIRMAN ARMIJO: Okay.
5 MR. CLIFFORD: And I can go back and look
6 at the strain.
7 CHAIRMAN ARMIJO: Yes, Paul, just make sure
8 I'm not just confusing myself. For the radial fall-off
9 curve ramp LHGRs, those take account of the trip, right,
10 or not? That's 112 percent overpower and that's as high
11 as it can go because of the trip.
12 MR. CLIFFORD: Correct. Right, you're
13 basing this on the assumption that these are global
14 changes in power, which the excess steam demands it would
15 be a global increase in power in that if a rod was
16 operating at five it would go to five times 112 percent
17 here for this case, and if it was operating at nine it
18 would go to nine times 112 percent. So you're saying the
19 trip function limits the, from a global power increase
20 --
21 CHAIRMAN ARMIJO: That's as bad as it can
22 get.
23 MR. CLIFFORD: Right. So and the next
24 investigation, as I mentioned, what we did was instead
25 of limiting the rod power I just iterated on power. I

1 iterated on power until I reached the 200 megapascal
2 lower limit and the 250 best estimate thresholds. And
3 this was to see how much margin was there between the
4 reactor trip setpoint and when I would expect cladding
5 hoop stress to achieve these threshold values.
6 So at a lower burnup you're looking at about
7 124 to 129 percent overpower before you achieve those
8 two. The lowest fraction you see is down at the highest
9 burnup. It's 118.4. That's the value to achieve the
10 200, but that's up at 125.5 at the best estimate
11 threshold.
12 So the purpose of this exercise was just to
13 kind of quantify what the margin is. I mean really the
14 purpose of this whole investigation is just to say, okay,
15 is there an immediate safety concern? Is there
16 something that jumps off the page that makes you say,
17 well, this is an issue? And I don't think there is.
18 I think if you look at the characteristics
19 of PWRs and if you look at the trip functions that are
20 currently in place, I don't think that you can dream up
21 a prolonged power event that's going to be of sufficient
22 magnitude, because it is limited by the trip functions,
23 to get cladding stress to the point where you would
24 achieve PCI through-wall cladding failure.
25 CHAIRMAN ARMIJO: Paul, and I would totally

1 agree with you if I agreed with that threshold number.
2 So it's like, you know, look at it from a strain point
3 of view and see if you still feel as confident about that.
4 But I think you've addressed the question
5 that I've been asking for a long time and many of us have
6 been asking very clearly, so I'm pleased. The only thing
7 I have reservations about is the issue of what's the right
8 threshold to use, and maybe stress is the right threshold
9 but what is the right value of that stress?
10 MR. CLIFFORD: Right. And in the
11 conclusions, you know, I try to address that. I mean
12 here, what we've done is we've kind of done a quick and
13 dirty vulnerability assessment. Is there a reason to
14 believe that there's an immediate safety problem, should
15 we take immediate actions? And I think the answer to
16 that is no.
17 I agreed, Sam, with everything you're
18 saying. I think we need to continuously revisit this.
19 And the Office of Research will continue to work with the
20 SCIP program, and they've got very focused research
21 activities planned that will hopefully come up with a
22 better failure threshold whether that's a certain strain
23 or it's a certain stress.
24 I mean this was just our first attempt at
25 this and, you know, that 200 megapascals, at least from

1 the data, looks like a very conservative failure
2 threshold since there was many calculated stresses that
3 didn't fail above 200. And the lowest failure one was
4 230.
5 So I think it was a conservative initial
6 attempt to kind of investigate susceptibility of PWRs.
7 And as I say in this bullet here, I think we should revisit
8 this. Once we get a better failure threshold we should
9 go back and repeat these calculations and see how it all
10 falls out.
11 The last bullet is, I think it's important
12 that the staff continue monitoring PWR designs and
13 operations to ensure that PCI doesn't, you know, become
14 a safety significant issue. And that in my mind would
15 mean, really, load following. If you started to see
16 plants using control rods to load follow, then now
17 they're putting themselves in the situation where the
18 BWRs used to be.
19 CHAIRMAN ARMIJO: Well, Paul, I totally
20 agree with that last bullet. But the design can change.
21 You know, if somebody came to you and says, Paul, I really
22 would like to get more UO2 in that fuel bundle and I want
23 to tighten up that gap another half mil, or I want to
24 increase the density of the pellet by another half
25 percent, you're eating into margin.

1 MR. CLIFFORD: Oh, absolutely.

2 CHAIRMAN ARMIJO: And if this isn't viewed
3 as part of the things you have to worry about in the
4 design, and if it's not in the Safety Review Plan to look
5 at it in a quantitative way, people will say well, I meet
6 the SAFDLs for PCMI, and that's not the way to do it.
7 That's not a good way to leave it.

8 MR. CLIFFORD: No, I agree. I just think
9 it's just not ready. We're not ready to go out there with
10 a new regulatory requirement that we've backfit on the
11 industry. We don't have a technical basis that's robust
12 enough to walk through that exercise.

13 CHAIRMAN ARMIJO: Well, I'm not proposing
14 that we backfit anything or a regulatory requirement.
15 But what I think should be done is that this failure
16 mechanism should be analyzed for itself and not, no
17 assumption should be made that some other criterion
18 protects you against this.

19 The BWRs, by and large, have a liner and so
20 they've got built-in PCI resistance. The PWRs, and when
21 we, you know, we're seeing now that they're extremely
22 mild as far as the kind of transients that can occur
23 before you trip. That's good news. And I mentioned the
24 other things that I always thought were a built-in
25 advantage of the Ps from the standpoint of PCI. So all

1 of that's working for it.

2 The only thing, and I think you're so close

3 to really nailing this thing by looking at the strains

4 from well characterized fuel tests, and just seeing if

5 FRAPCON would predict the kinds of strains that the BWR

6 rods failed at.

7 MR. CLIFFORD: Right.

8 CHAIRMAN ARMIJO: And if you had that then

9 I would say, it isn't a leap of faith to say that code

10 using the same strain criteria would tell me where I am

11 on the PWR transient. And you've got a tool.

12 MR. CLIFFORD: Right.

13 CHAIRMAN ARMIJO: It may be empirical, but

14 it's still better than what we've got.

15 MR. CLIFFORD: Yes, I just wonder if even

16 if you plotted out all that data whether you wouldn't see

17 the same degree of overlap in all the results.

18 CHAIRMAN ARMIJO: Have you looked at the

19 BWR data?

20 MR. CLIFFORD: Well, for this exercise we

21 excluded the BWR data.

22 CHAIRMAN ARMIJO: No, I'm just saying --

23 MR. CLIFFORD: There's liner data mixed in

24 there and --

25 CHAIRMAN ARMIJO: No, but to really show

1 the FRAPCON --

2 MR. CLIFFORD: Oh no, FRAPCON has been

3 qualified against a pretty extensive database and, you

4 know --

5 CHAIRMAN ARMIJO: Yes, but if it's good it

6 should be able to predict the failure strain of the BWR

7 ramp tests that aren't confounded by the billions of

8 variables that Patrick pointed out for the PWR data. And

9 even there, if he just sorts out the variables, you'll

10 have smaller datasets that are sound. So anyway, I'll

11 leave that as a plea, request, urge that you think about

12 that.

13 MR. CLIFFORD: Okay.

14 CHAIRMAN ARMIJO: Okay, any other

15 comments, questions? Ron?

16 MR. BALLINGER: I think I'm saying what you

17 said. You can reduce both the PWR and the BWR to strains

18 on the cladding, and you can then forget about whether

19 it's a BWR or PWR, it's just strain on the cladding.

20 So the maneuvering restrictions on the BWR

21 work, how close are you to the BWR maneuvering

22 restrictions with this? My suspicion is that you're way

23 away from them. In other words, the strains and strain

24 rates on the cladding that you calculate here, don't come

25 close to the BWR maneuvering restrictions. In other

1 words, the strains that are not failing BWR, even though
2 there's a liner.
3 MR. CLIFFORD: I would say that's true in
4 the sense that you're going to get a pretty severe, or
5 you could, you have the opportunity to get a very severe
6 local power increase when you're moving control blades
7 versus these relatively slow increases to secondary
8 steam demand which is creeping up to 109 percent. I mean
9 you certainly see more than a nine percent increase in
10 local power density when you move a blade.
11 MR. CLIFFORD: So if you were to take
12 non-barrier cladding, PWR non-barrier cladding where you
13 don't fit, where does the strain for non-failure of BWR
14 cladding compare to this?
15 MR. BALLINGER: Right. It's something to
16 look at. And there are plants that do have non-barrier
17 --
18 MR. BALLINGER: Yes.
19 MR. CLIFFORD: -- and they have the
20 operating restrictions. They're certainly different
21 than the barrier fuel operating restrictions, but
22 they're there.
23 CHAIRMAN ARMIJO: Okay. Dick?
24 MEMBER SKILLMAN: Thank you, no.
25 CHAIRMAN ARMIJO: Dr. Shack?

1 MEMBER SHACK: No, it was very interesting.
2 You know, as I say, I think the data is going to be
3 confusing. You know, I think I'm fairly comfortable
4 with just looking at that data where you at least sort
5 of have a 50 percent probability of failure at the kind
6 of stresses we were looking at in the tests. They're
7 still well below those, so whether I could rigorously
8 show that they're a threshold, I'd certainly say that
9 it's a fairly low probability that I'm going to get a
10 failure with those kinds of stresses. So, you know, it's
11 enough to give me kind of a warm, fuzzy feeling about it.
12 And again, at this point I think you'd need
13 a lot more data before you could ever have requirements
14 on it just because we don't have a model that I think could
15 be universally agreed upon.
16 CHAIRMAN ARMIJO: Joy? Comments?
17 MEMBER REMPE: Again, I appreciated the
18 discussion because I've learned a lot. Out of
19 curiosity, I'm still wondering about the data that we saw
20 during the closed session, and if perhaps it was obtained
21 under a lot more severe conditions.
22 And so if you went -- or again, I don't quite
23 have that comparison so it would be good to see how it
24 compares to the actual transients that are in the PWRs,
25 because I think you might want to be ruling out some of

1 the data and you might end up with a different threshold
2 one way or the other.
3 CHAIRMAN ARMIJO: Okay, some left. Well,
4 first of all, I would like to thank Paul. I think you
5 did a good service. You answered the question. You
6 addressed the question exactly what the concern was.
7 And, you know, as I said, it's more than a quibble is that,
8 you know, Paul presented a ton of data with too many
9 variables so it leads one to believe that there's no hope.
10 And that's not the case.
11 I've analyzed a lot of PCI data, and you just
12 have to limit the variables. And if you start, do tests
13 with the control of the number of variables, you find it's
14 not so, it's not a deterministic thing. Every test done
15 to a certain power won't result in failure, but the higher
16 you go in power the more consistent it gets. Everything
17 fails.
18 So if you don't have enough good PWR
19 datapoints into the power range of interest to you in the
20 AOO, let's say you're ten kilowatts per foot or whatever
21 it is at the maximum that you calculate for that radial,
22 then you either have to have more data or PWR rods, or
23 you can use FRAPCON but use the data that's available in
24 the BWR rods.
25 It's all zircaloy and it's all, you know,

1 it's the same phenomena going on. And I think you could
2 actually calibrate this thing using a failure strain
3 criteria, which is the only thing you can measure rather
4 than failure stress. And then you'd have data to support
5 your threshold if you want to do it. And it seems to me
6 that's a real direct way to show that you can predict PCI
7 failure.
8 MR. CLIFFORD: But I mean these balloon
9 rupture, you know, in an iodine environment, those tests
10 that they're doing, that they have done and they plan on
11 expanding, I mean that will give you certainly a
12 measurable hoop stress.
13 CHAIRMAN ARMIJO: Well, you know, I'll tell
14 you, and I know it sounds like perhaps we're tooting work
15 that was done many years ago, but pressurized tube tests,
16 expanding mandrel tests were done in the GE test program
17 in developing barrier fuel, and they serve a purpose.
18 But they're nothing close to -- there are
19 parallels between their behavior and the behavior of a
20 rod in a ramp test in a real reactor. And so you can't
21 calibrate a model or a fuel rod from just an expanding
22 mandrel test or a pressurized tube test. I think you'll
23 get the wrong answer.
24 So if you have a good set of ramp test data
25 that's worth its weight in gold. And believe me, it's

1 millions of dollars that went into generating those ramp
2 tests data and they really would be valuable to use.
3 Okay, well, look, I want to thank everybody
4 for their patience, and we're actually a little bit ahead
5 of schedule. And I'm very pleased that we've made a lot,
6 I think the staff's made a lot of progress and I know
7 you've put effort into this, Paul, and I appreciate that.
8 With that, the meeting is adjourned. Thank you.
9 (Whereupon, the foregoing matter went off
10 the record at 4:19 p.m.)
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Risk of Fuel Cladding Failure Due to PCI/SCC During PWR AOOs

ACRS Materials, Metallurgy & Reactor Fuels Subcommittee Meeting

June 17, 2013

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Agenda:

1. Current Regulatory Practice
2. Studsvik Cladding Integrity Program – PCI/SCC Research (Closed)
3. PWR Operating Conditions & AOO Overpower Scenarios
4. FRAPCON-3.4 Calculations
5. Conclusions

PCMI versus PCI

- Fuel rods are manufactured with a prescribed gap between the fuel pellet and the cladding inside diameter. During reactor operation, fuel pellet thermal expansion, irradiation-induced fuel swelling, and irradiation-assisted cladding creep-down bring about a reduction in the size of this gap and eventually hard contact between the fuel pellet and the cladding. This is referred to as pellet-to-cladding mechanical interaction (PCMI).
- Any increase in fuel rod power will result in a proportional increase in fuel temperature which, in turn, increases fuel rod pellet diameter (due to thermal expansion) and may produce cladding stresses and strain.
  - Resulting cladding stress/strain depend on initial gap size, prior operating history (e.g., fuel burnup), and magnitude of local power change.
- Cladding stress combined with the aggressive chemical agents present in fission products (e.g., iodine) may lead to cladding failure via stress corrosion cracking (referred to as PCI or PCI/SCC).

Current Regulatory Practice – GDC10

10 CFR 50 Appendix A, General Design Criteria

Criterion 10—Reactor design. The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Current Regulatory Practice – SRP

In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

1) The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
2) Fuel system damage is never so severe as to prevent control rod insertion when it is required,
3) The number of fuel rod failures is not underestimated for postulated accidents, and
4) Coolability is always maintained.

Current Regulatory Practice – SRP

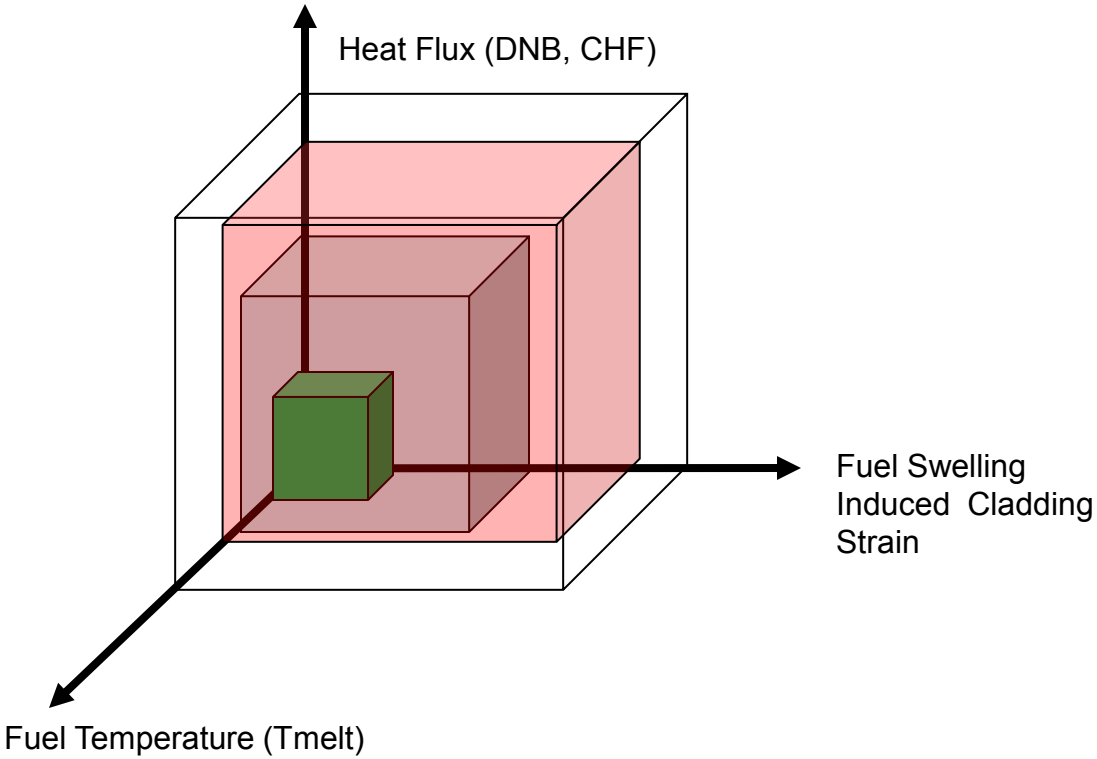
In accordance with SRP Section 4.2, SAFDLs defined based on all known degradation mechanisms. Current fuel rod SADFLs include:

1) Cladding-to-coolant heat flux (DNB, dryout)
2) Fuel temperature (fuel melt)
3) Fuel swelling (cladding strain)
4) Rod internal pressure (clad liftoff)
5) Creep collapse
6) Stress, strain, and fatigue
7) Fuel rod growth
8) PCI/SCC (BWR power maneuvering)

Additional design requirements associated with fuel bundle design.

Preventing AOO Fuel Failures

- NPPs rely on fuel design features, operating procedures, LCOs (e.g., initial thermal margin), automatic system actuations, and manual response to preclude fuel failure during normal operation and AOOs.
- UFSAR AOO simulations based on conservative analytical models, limiting initial conditions, most unfavorable, allowable system response, and high confidence safety limits (e.g., 95/95 DNBR).



- Would consideration of PCI/SCC cladding failure during AOOs require further restrictions on NPP design and operations?

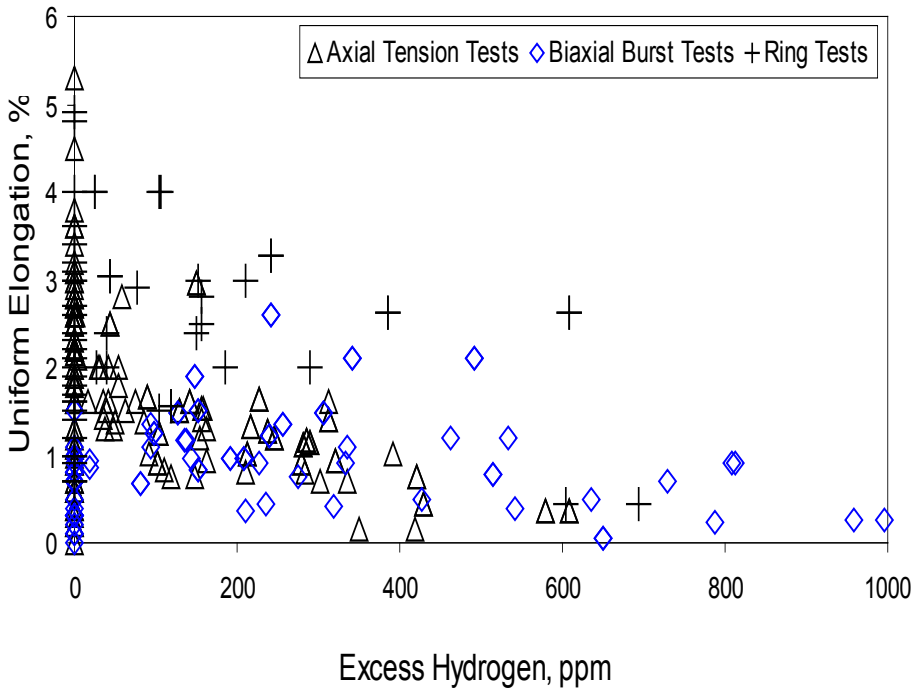
Current Regulatory Practice – GE14E

Text from staff's SE for GE14E fuel assembly design:

SAFDLs on fuel rod cladding strain and fuel centerline melting are employed to preclude fuel rod cladding failure due to PCMI during rapid over-power AOOs. However, as described in SRP 4.2, these design limits may not provide sufficient protection to preclude fuel cladding failure due to stress corrosion cracking (PCI/SCC) under certain sustained cladding loading conditions. In response to an RAI regarding the PCI/SCC resistance of GE14E fuel (RAI 4.8-12 of Reference 3), GEH provide results from past and recent ramp test programs which are applicable to GE14E's barrier design. This information shows that margin exists between current operating limits and an empirically-based lower failure threshold such that PCI/SCC failures would not occur during power maneuvering. Furthermore, GE's barrier cladding design has been proven to reduce PCI/SCC susceptibility during both power maneuvering and AOO type scenarios. As a result, PCI/SCC fuel cladding failure is unlikely during any AOO scenario which experiences a sustained power excursion and does not already predict fuel rod failure due to violating other SAFDLs (e.g., MCPR, cladding strain, centerline melt). Hence, there is reasonable assurance that fuel cladding failure would not be underestimated.

PCMI - Cladding Strain Capability

- Fuel vendors are responsible for defining cladding strain SAFDL (e.g., 1.0% plastic+elastic) for their zirconium alloys and providing supporting empirical data on irradiated cladding segments.
- Cladding strain capability is a function manufacturing parameters (e.g., alloy composition as it related to corrosion resistance, heat treatment as it related to hydride orientation and distribution), neutron fluence, and temperature.



PCI/SCC - Sensitivities

- Observations from 200+ Studsvik ramp tests on PWR fuel rod segments at PWR pressure and temperature will be discussed in closed session.

PWR Operating Conditions & AOO Overpower Scenarios

PWR Operating Experience – PCI/SCC

- Operating experience has shown that BWRs are susceptible to PCI/SCC while performing plant maneuvering with control blades.
  - Previously controlled portions of BWR fuel rods experience prolonged periods of high power and high cladding stress following the withdrawal of a high worth control blade.
- Operating experience has shown that PWRs are <u>not</u> susceptible to PCI/SCC during normal operations.
  - Limited number of PCI/SCC cladding failures experienced due to missing pellet surface (MPS).
  - Problem solved via improved manufacturing specifications and inspection.

PWR NSSS Designs

- PWR NSSS designs currently operating:

6	Babcock & Wilcox 2-loop, once-through SGs
12	Combustion Engineering 2-loop, u-tube SGs
5	Westinghouse 2-loop, u-tube SGs
13	Westinghouse 3-loop, u-tube SGs
29	Westinghouse 4-loop, u-tube SGs

PWR Operating Conditions

- Reactivity control:
  - BWRs rely on control blade insertion to hold down excess reactivity.
  - PWRs employ boric acid dissolved in RCS coolant to hold down excess reactivity.
- Control rod insertion:
  - BWRs operate with inserted control blades. These high worth blades may be individually inserted/withdrawn.
  - PWRs normally operate with all-rods-out (ARO). Control rods move within assigned banks and insertion restricted via TS/COLR power dependent insertion limits (PDILs).

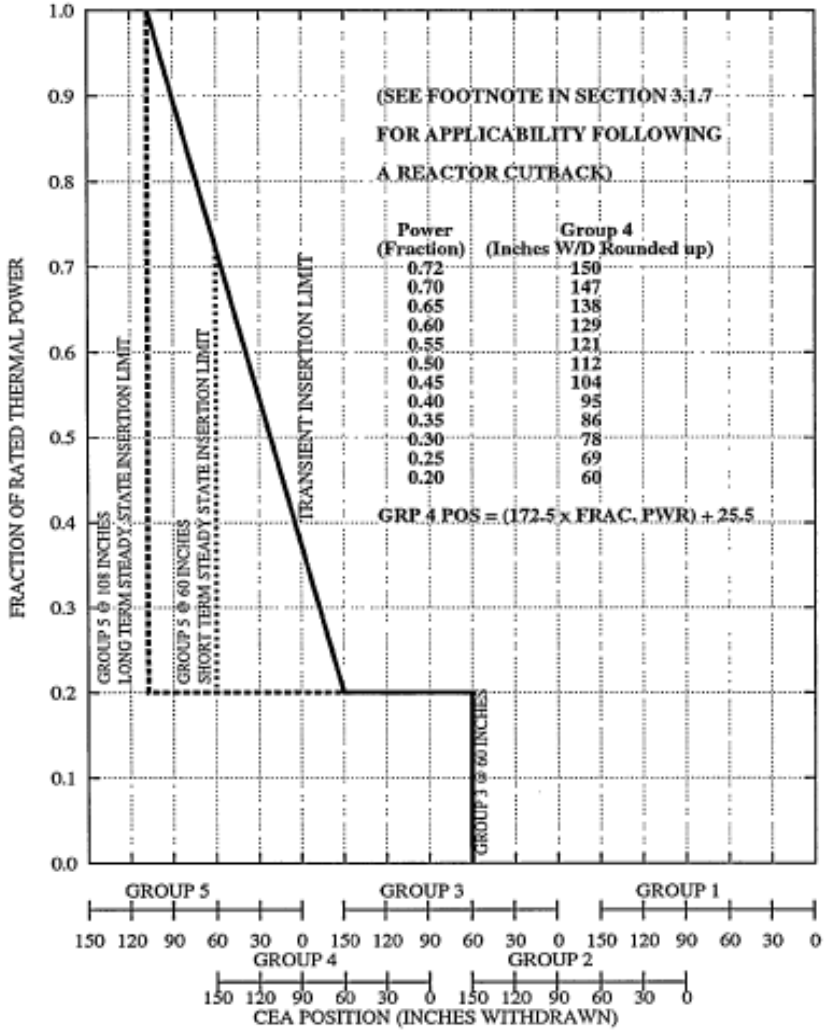
Typical PWR TS/COLR PDIL

PVNGS UNIT 2 CORE OPERATING LIMITS REPORT Revision 8

FIGURE 3.1.7-1

CEA INSERTION LIMITS VERSUS THERMAL POWER

(COLSS IN SERVICE)

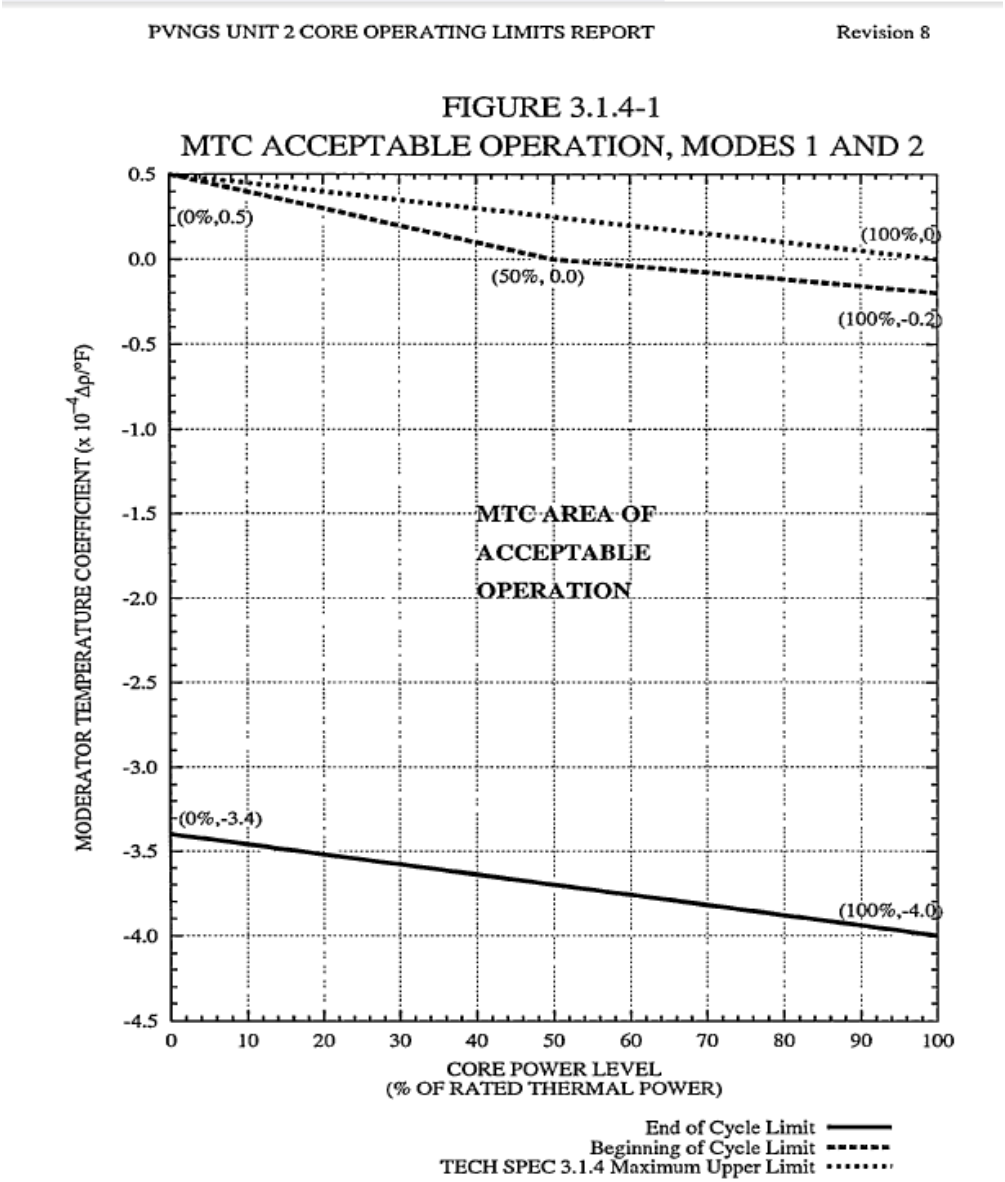


PWR Reactor Kinetics

- In addition to a negative fuel temperature coefficient (FTC), PWRs operate with a negative moderator temperature coefficient (MTC).
  - Due to high concentration of soluble boron at the beginning of a new operating cycle, MTC may be slightly positive at low power, but becomes negative prior to reaching full power.
- Due to negative temperature reactivity feedbacks, PWR are very stable with respect to responding to increasing or decreasing power scenarios.
  - Any increase in reactor power without a proportional increase in secondary demand will promote increasing reactor coolant temperature which, in turn, results in the addition of negative reactivity.

Reactor power follows secondary steam demand

Typical PWR TS/COLR MTC Limit



PWR Reactor Protection System

- For an AOO overpower scenario initiated from HFP conditions, PWRs rely on High Neutron Flux / Variable High Power Trip safety-grade reactor trip functions to provide a timely reactor trip to protect against violation of SAFDLs.
- Additional PWR reactor trip functions which protect against overpower scenarios (and likely to provide a trip prior to high flux setpoint) include Overpower ΔT (Westinghouse and MHI) and Low DNBR (CE-Digital and AREVA EPR).

High Power Trip Setpoints

Design	Allowable Value	Source
Westinghouse	≤ 110.0% RTP	McGuire TS Table 3.3.1-1
B&W	≤ 105.5% RTP	Oconee TS Table 3.3.1-1
MHI US-APWR	109.0% RTP	DCD Table 7.2-3 (Rev.3)
AREVA EPR	105.0% RTP	DCD Table 15.0-7 (Rev.4)
CE (Analog)	< 107.0% RTP	Calvert Cliffs TS Table 3.3.1-1
CE (Digital)	≤ 111.0% RTP[1]	PVNGS TS Table 3.3.1-1
	< 110.0% RTP[2]	PVNGS UFSAR Table 7.2-1A

[1] RPS analog variable over power trip (VOPT), band set at ≤ 9.9% RTP above power, no compensation.
[2] CPCS digital VOPT, band set at ≤ 8.0% RTP above power, dynamic compensation.

Safety Analysis Methodology

- A review of several PWR UFSARs reveal that analysts perform parametric studies to determine the limiting combination of initial conditions and assumptions to either delay or avoid reactor trip(s) or maximize the over-shoot of the trip setpoint (to maximize power).
  - This strategy yields conservative analyses with respect to limiting margin to DNB, Tmelt, and cladding strain SAFDLs for AOOs and maximizing fuel failure for postulated accidents
- PCI/SCC crack propagation requires both cladding stress and time.

Does an AOO overpower scenario exist which exhibits a prolonged power excursion of sufficient magnitude to experience PCI/SCC cladding failure?

Survey of PWR AOOs

Event	Local Power Increase	Global Power Increase
Decrease in Feedwater Temperature	N	Y
Increase in Feedwater Flow	N	Y
Increase in Main Steam Flow	N	Y
IOSGADV	N	Y
SBCS Malfunction	N	Y
Loss of Load / Turbine Trip	N	N
Loss of Condenser Vacuum	N	N
Loss of Forced Flow / LOAC	N	N
MSIV Closure	N	N
Inadvertent Bank CEA Withdrawal	Y	Y
Loss of Feedwater Flow	N	N
Inadvertent Boron Dilution	Lower Mode	
Control Rod Drop	Y	N
PPCS Malfunction	N	N
Startup of an Inactive RCP	Lower Mode	
CVCS Malfunction	N	N
ASGT	Y	N

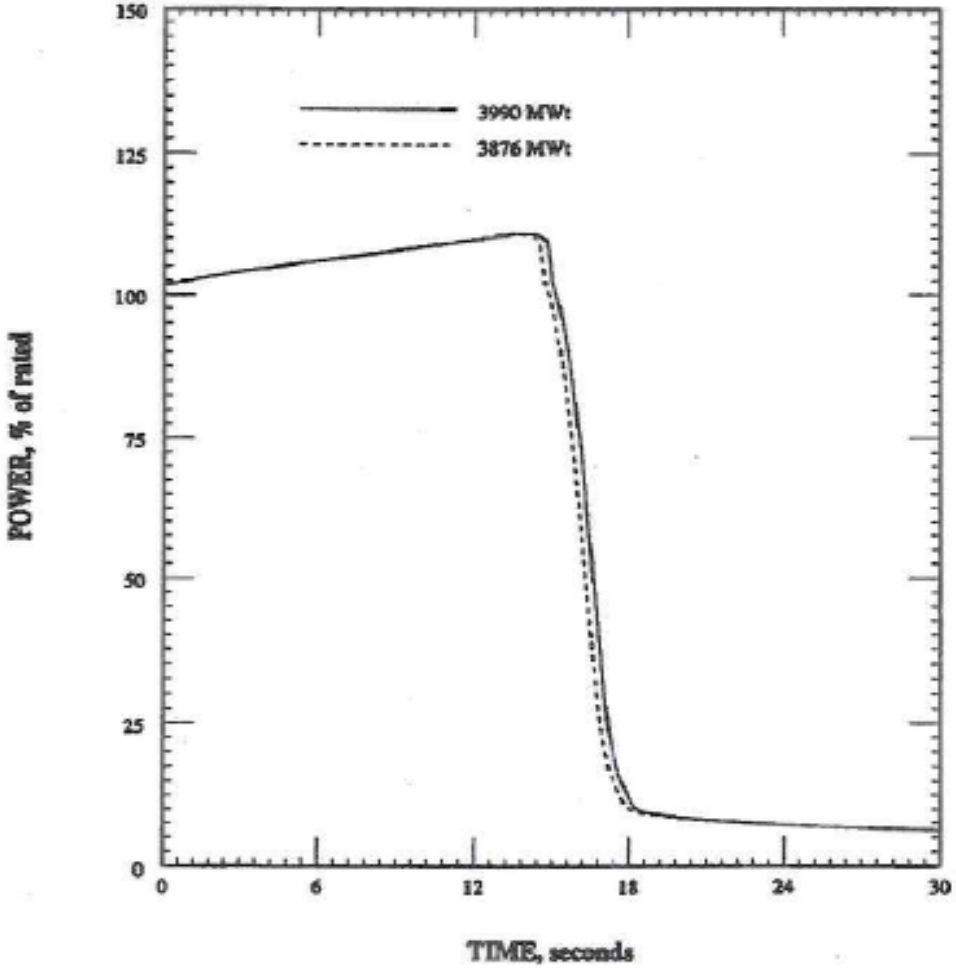
Bank CEAW

- The limiting CEAW scenario maximizes the power excursion (i.e. over-shoot) by selecting the maximum bank worth, least negative MTC, least negative FTC, and minimum delayed neutron fraction (β).
- It is possible to identify a CEAW scenario (e.g., low bank worth) which avoids a quick reactor trip.
- However, with no increase in secondary steam demand, reactor power will eventually trend down due to temperature feedback (following 8-10 second loop transition time).

Bank CEAW (cont.)

- Based upon the following items, susceptibility to PCI/SCC cladding failure is low.
  - ARO operation means event probability is very low.
  - Bank worth (Δρ/in) and resulting power excursion and power peaking minimized by TS/COR rod insertion limits.
  - Negative temperature reactivity coefficients (FTC, MTC) resist power excursion.
  - Reactor Protection System High Power trip function (e.g. 110% high flux) limit magnitude and/or duration of power excursion. Additional trip functions include Overpower ΔT (Westinghouse and MHI) and Low DNBR (CE-Digital and AREVA EPR).
  - Several control room alarms available to alert operators.

Bank CEAW



Source: PVNGS UFSAR Figure 15.4.2-1

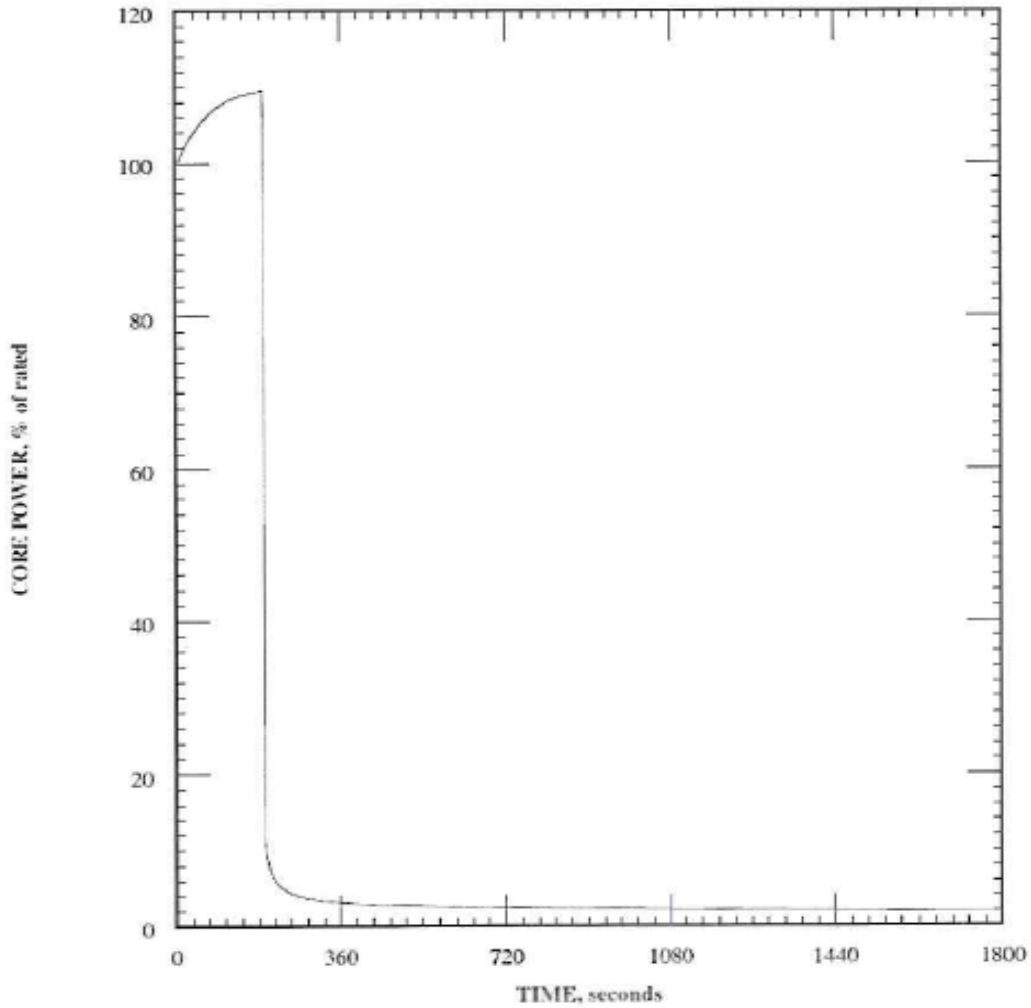
Excess Steaming Events

- Since reactor power follows secondary steam demand, it seems that scenarios which promote an increase in secondary heat removal are likely candidates for this PCI/SCC investigation.
- The AOO survey reveals that 5 AOOs in this category all exhibit a global power increase.
- Depending on the magnitude of excess heat removal and MTC, a wide range of system responses is possible.
  - With the right combination of excess heat removal and MTC, a prolonged overpower event is possible.

Excess Steaming Events (cont.)

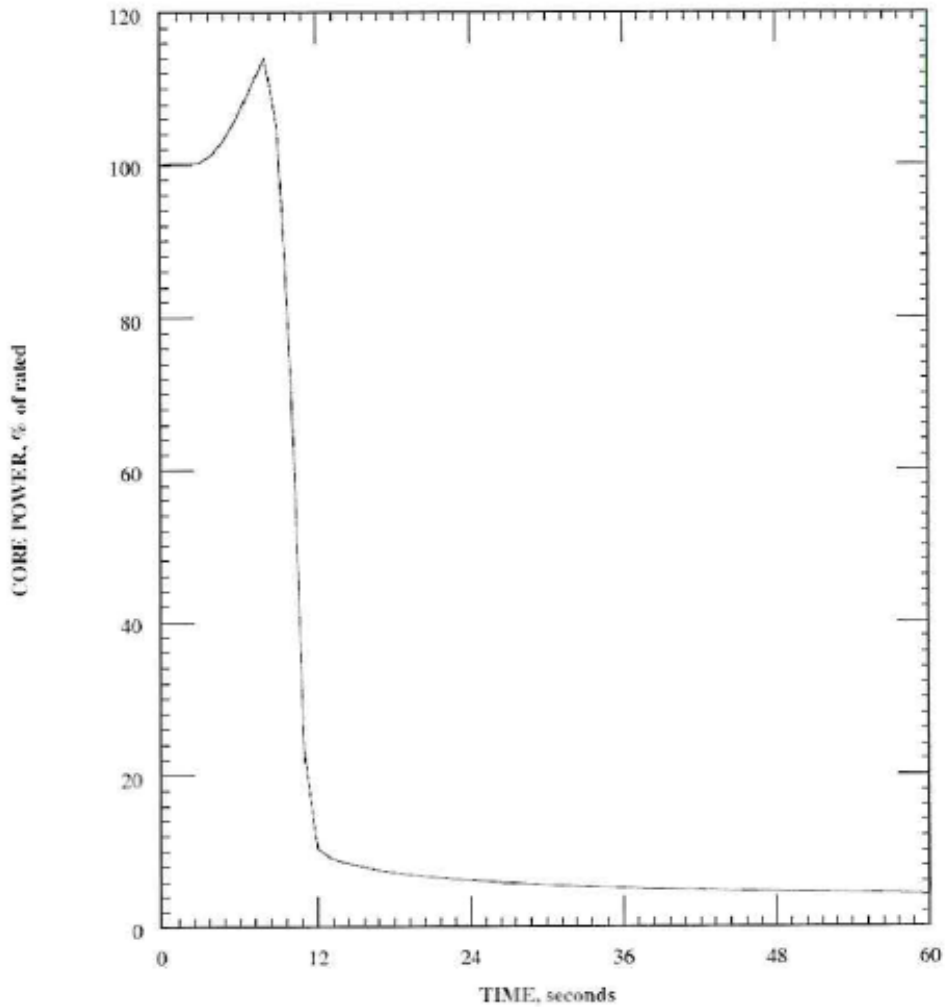
- Based upon the following items, susceptibility to PCI/SCC cladding failure is low.
  - Reactor Protection System High Power trip function (e.g. 110% high flux) and Low SG Pressure limit magnitude and/or duration of power excursion. Additional trip functions include Overpower ΔT (Westinghouse and MHI) and Low DNBR (CE-Digital and AREVA EPR).
  - Several control room alarms available to alert operators.
  - Balance of plant equipment protection trips.

IOSGADV



Source: PVNGS UFSAR Figure 15.1.4-6

SBCS Malfunction



Source: PVNGS UFSAR Figure 15.1.3-4

FRAPCON-3.4 Cladding Stress Calculations

FRAPCON-3.4 Calculations

Two different sets of calculations were performed to address the following questions:

1. Are current TS HPT setpoints sufficient to prevent PCI/SCC cladding failure during core-wide AOOs?
   a) Is the predicted cladding hoop stress during a prolonged AOO overpower scenario of sufficient magnitude to promote PCI/SCC cladding failure?

2. What is the magnitude of power excursion necessary to reach a predicted cladding hoop stress equivalent to the empirically-based, draft failure threshold(s)?

Inputs and Assumptions

- Power excursion:
  - A rapid power excursion which over-shoots HPT setpoint will yield higher cladding stress; however its time duration will not be sufficient for crack propagation.
  - As a result, prolonged power excursion must remain below HPT setpoint. An instantaneous ramp to 112% of initial rod power is assumed.
- Rod design and power history:
  - Limiting 2nd cycle fuel rod from prior EPU study.
  - Limiting 3rd cycle fuel rod from prior EPU study.
  - Bounding radial fall-off curve (Reload Checklist, Fr vs. BU)
- Cladding failure threshold:
  - 200 MPa (lower bound)
  - 250 MPa (best-estimate)

Investigation #1 – Ramp to 112%

- Power ramps simulating an AOO overpower scenario (e.g., IOSGADV) were added at various burnup steps to the three base power histories.
- The maximum predicted cladding hoop stress was 164.2 MPa for the SPH7 UO2 fuel rod.
  - With a power of 7.75 KW/ft at 66.9 GWd/MTU, this represents a bounding power history.
- Investigation #1 shows that predicted cladding hoop stress remains below the lower bound 200 MPa and best-estimate 250 MPa failure thresholds for a prolonged overpower scenario.

These calculations suggest that existing TS HPT setpoints are sufficient to prevent PCI/SCC.

Investigation #1 – Ramp to 112%

Time Step	Local Burnup (GWd/MTU)	Initial LHGR (KW/ft)	Ramp LHGR (KW/ft)	Initial Hoop Stress (MPa)	Final Hoop Stress (MPa)
Limiting 2nd Cycle UO2 Fuel Rod					
2	0.2	8.01	8.97	-67.4	-65.4
15	20.7	7.55	8.46	-58.4	-55.5
29	39.7	6.58	7.49	-49.9	-4.4
38	48.7	6.43	7.21	64.3	120.2
Limiting 3nd Cycle UO2 Fuel Rod					
38	40.5	5.55	6.22	58.3	102.7
46	50.9	5.85	6.55	78.5	135.9
55	58.7	5.75	6.48	107.8	160.5
Radial Fall-Off Curve					
SPH3	31.2	9.55	10.7	15.4	104.1
SPH4	40.4	9.43	10.6	29.3	122.6
SPH5	48.9	9.39	10.5	39.5	138.9
SPH6	58.3	8.71	9.7	48.9	145.9
SPH7	66.9	7.75	8.8	73.6	164.2

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Investigation #2 – Iterated Power

- Rod power was iterated at various burnup steps until cladding hoop stress achieved the draft failure thresholds.
- The FRAPCON-3.4 calculations show that a prolonged power excursion greater than 118.4% (lower bound) and 125.1% (best-estimate) is necessary to exceed their respective PCI/SCC failure threshold.

These calculations suggest that existing TS HPT setpoints are sufficient to prevent PCI/SCC.

Investigation #2 – Iterated Power

Time Step	Local Burnup (GWd/MTU)	Iterated Power Ramp	
		200 MPa KW/ft / (%)	250 MPa KW/ft / (%)
SPH3	31.2	11.9 (124.3%)	12.4 (129.6%)
SPH4	40.4	11.5 (122.0%)	12.0 (126.7%)
SPH5	48.9	11.2 (119.7%)	11.8 (125.1%)
SPH6	58.3	10.0 (123.7%)	10.5 (130.0%)
SPH7	66.9	9.2 (118.4%)	9.7 (125.5%)

Conclusions

- Due to design features and operating restrictions, PWRs are not susceptible to PCI/SCC cladding failure during normal operation. This conclusion is backed by operating experience.
- With respect to vulnerability during AOOs, FRAPCON-3.4 calculations suggest that existing safety-grade reactor trip functions are adequate to provide reasonable assurance against PCI/SCC cladding failure.
- NRC staff will continue to participate in international research activities investigating PCI/SCC.
  - This assessment will be re-visited once failure thresholds are better defined.
- NRC staff will continue monitoring PWR design and operations (e.g. load following) to ensure PCI/SCC does not become a safety significant issue.