UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Title: BRIEFING ON PRA IMPLEMENTATION PLAN PUBLIC MEETING

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2 NUCLEAR REGULATORY COMMISSION

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4 BRIEFING ON PRA IMPLEMENTATION PLAN

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6 PUBLIC MEETING

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9 Nuclear Regulatory Commission

10 Commission Hearing Room

11 11555 Rockville Pike

12 Rockville, Maryland

13

14 Wednesday, October 15, 1997

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16 The Commission met in open session, pursuant to

17 notice, at 10:05 a.m., the Honorable SHIRLEY A. JACKSON,

18 Chairman of the Commission, presiding.

19

20 COMMISSIONERS PRESENT:

21 SHIRLEY A. JACKSON, Chairman of the Commission

22 GRETA J. DICUS, Member of the Commission

23 EDWARD McGAFFIGAN, JR., Member of the Commission

24 NILS J. DIAZ, Member of the Commission

25

1 STAFF AND PRESENTERS SEATED AT COMMISSION TABLE:

2 KAREN D. CYR, General Counsel

3 JOHN C. HOYLE, Secretary

4 MARGARET V. FEDERLINE, NMSS

5 SCOTT F. NEWBERRY, NRR

6 L. JOSEPH CALLAN, EDO

7 ASHOK C. THADANI, OEDO

8 THOMAS L. KING, RESEARCH

9 PATRICK W. BARANOWSKI, AEOD

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

2 CHAIRMAN JACKSON: Good morning. I'm pleased to
3 welcome members of the staff to brief the Commission on the
4 status of the PRA Implementation Plan.
5 The PRA Implementation Plan was first issued in
6 August 1994. The Plan is intended to be a management tool
7 that will help to ensure the timely and integrated
8 agency-wide use of PRA methods and technology in the
9 agency's regulatory activities.
10 The last written update on the status of
11 activities in the PRA Implementation Plan was received
12 recently by the Commission. The Commission was last briefed
13 on the Plan in May of this year.
14 During today's briefing, the staff will cover its
15 recent accomplishments -- status of key activities,
16 responses to SRM's, and future activities. These new
17 activities include the development of standards for PRA and
18 the evaluation of the need to revise the Commission's safety
19 goal policy statement.
20 The staff's recent accomplishments -- and I'm
21 taking Joe's thunder, probably -- include the issuance of
22 draft risk-informed regulatory guidance for inservice
23 inspection for comment. A public workshop to discuss public
24 comment on these documents is being planned for later this
25 year. I am my fellow commissioners are looking forward to

1 your briefing today, and I understand that copies of the
2 viewgraphs are available at the entrances to the meeting.
3 So, if none of my colleagues have any comments they wish to
4 make, Mr. Callan, please proceed.
5 MR. CALLAN: Thank you, Chairman. Ashok Thadani,
6 to my right, who is the deputy EDO for regulatory
7 effectiveness, will lead the staff's discussion this
8 morning, but before I turn the meeting over to him, let me
9 introduce the other members at the table.
10 We have a diverse group of executives at the table
11 representing all the large program offices -- to my far
12 left, Margaret Federline, representing the Office of Nuclear
13 Material Safety and Safeguards; Scott Newberry, representing
14 the Office of Nuclear Reactor Regulation. I've already
15 introduced Ashok. To his right, Tom King, representing the
16 Office of Nuclear Regulatory Research; and then, Pat
17 Baranowski, representing the Office of AEOD.
18 With that, Ashok.
19 MR. THADANI: Thank you, Joe. Good morning.
20 CHAIRMAN JACKSON: Good morning.
21 MR. THADANI: Well, Chairman, as you know, the
22 most prominent activity underway in the Implementation Plan
23 continues to be the development of regulatory guidance
24 documents -- that is the Reg Guides and Standard Review
25 Plans.

1 In August, we held a workshop. I attended that
2 workshop -- part of the workshop. It was a three-day
3 workshop. It was very well attended. It was very lively;
4 there was a great deal of give-and-take. We received 30
5 sets of written comments on the guidance documents. Some of
6 the comments are quite significant, some significant in
7 terms of technical issues, as well as policy matters.
8 An example of a policy issue that has been raised
9 is concerned with having so-called tight limit of using a
10 core damage frequency of 10^{-4} per reactor year in terms of
11 considering any further small increases in risk.
12 There were questions along the lines of
13 clarification as amendments come in with varying impact on
14 core damage frequency, what was meant by different level of
15 analyses, as well as different level of management
16 attention, as we discussed in the past.
17 There were also substantial questions in the area
18 of uncertainties -- to what extent the detailed uncertainty
19 analyses need to be conducted for very minor or very small
20 changes in, let's say, core damage frequency.
21 We're analyzing these comments, and we're in fact
22 planning to discuss our initial thoughts on these issues
23 with the Advisory Committee next week. We would expect --
24 as Tom is going to summarize some of these issues, we would
25 expect to come back to the Commission on the policy issues

1 for guidance.

2 CHAIRMAN JACKSON: Let me ask you a question that
3 occurs to me immediately. You mention this issue of this
4 tight upper limit vis-a-vis the core damage frequency -- you
5 know, 10^{-4}. At the same time -- and I'm only looking aqt
6 what I read in the trade press -- there's discusion about
7 the industry having a perspective of wanting to use PRA's as
8 they are.

9 The third piece is, when I've pressed the staff in
10 meetings that have been going on since I've been here, about
11 the PRA's -- what you can say about them, the quality of
12 them, et cetera, et cetera, et cetera, et cetera -- there
13 has been some kind of squishiness and indeterminacy, and
14 there was some issue about PRA's that were graded, you know,
15 one, two, three, or something.

16 And so, the real question that I have is, frankly,
17 this -- that I think, in terms of your bringing a policy
18 issue to the Commission that I think has to be addressed, if
19 there's some variability in the PRA's, either in terms of
20 the methodologies or assumptions, et cetera, or fundamental
21 quality of them, that tracks with this issue of what kind of
22 limits or how much flexibility there can be in limits, et
23 cetera, et cetera, et cetera, that if you don't get at that
24 -- okay? -- and somebody says, "Well, my core damage
25 frequency is 10^{-5}," and if you did the calculation another

1 way, maybe it would be 10^{-3}, and if you did it another way,
2 maybe it would be 10^{-6}.
3 What are you doing about that? I have a question
4 about that. How do you handle that kind of issue in getting
5 at this? Because there's an assumption underlying all of
6 this that the numbers, as presented, have meaning.
7 MR. THADANI: Yes.
8 CHAIRMAN JACKSON: And that's a question.
9 MR. THADANI: Yes, clearly. And, in fact, that
10 was also one of the issues that was debated, which is the
11 guidance that we have in a NUREG document in terms of
12 quality for risk assessment.
13 CHAIRMAN JACKSON: The statement has been made
14 that you're looking for a gold-plated PRA.
15 MR. THADANI: Yes. There is concern that the risk
16 assessments don't need to be of that quality, and the
17 comment was made that perhaps the use of PRA or the quality
18 of the PRA should be driven by the application.
19 We have some thoughts on those issues in terms of
20 if the change has a truly negligible estimate impact on core
21 damage frequency -- we could define negligible - and, for
22 that change, does one need to go through a detailed
23 uncertainty analysis, for example.
24 It may be that that's not necessary, hut that
25 these are the issues that the industry has raised. We're

1 looking at them. We haven't really come to any conclusions
2 on that.
3 CHAIRMAN JACKSON: Right, but I think you have to
4 be thinking about these things. I mean the issue is, if you
5 have variability in the PRA's, in the quality of them or how
6 they're done, and getting straight at this issue of
7 uncertainty, because I've raised the issue in the past --
8 MR. THADANI: Yes.
9 CHAIRMAN JACKSON: -- the question is how much
10 variability and what degree of uncertainty, how much can be
11 tolerated for which regulatory use? Because I think it
12 tracks into some of the legal questions that arise.
13 MR. THADANI: Yes, it does.
14 CHAIRMAN JACKSON: And what's the difference
15 between the use of PRA in a risk-informed framework as
16 opposed to what I think some of the legal analysis has
17 focused on, which is risk-based?
18 MR. THADANI: That's right.
19 CHAIRMAN JACKSON: We're talking a risk-informed
20 framework, and how do these questions play into that?
21 MR. THADANI: Yes, indeed.
22 CHAIRMAN JACKSON: And I'm saying that, if you
23 don't address those, then don't send the paper, because I
24 think you're going to have to address these.
25 MR. THADANI: We would intend to address them.

1 Those are clearly the central issues. And I think, as you
2 have noted, it is significant to make sure that we are
3 talking about risk-informed and not risk-based, because
4 risk-based -- as you have noted, and it's pretty clear -- is
5 truly relying on numerical analyses almost as a central
6 basis. Our guidance documents don't really do that. I
7 think, in any case --
8 CHAIRMAN JACKSON: Sorry. I didn't mean to
9 preempt anything you were going to talk about.
10 MR. THADANI: I think, instead of my taking any
11 more time, it's better to just jump right into the issues.
12 CHAIRMAN JACKSON: Well, I think this commissioner
13 wants to jump first.
14 COMMISSIONER McGAFFIGAN: I just wanted a
15 clarification question. On the 10^{-4} --
16 MR. THADANI: Yes.
17 COMMISSIONER McGAFFIGAN: -- core damage
18 frequency, the upper limit, is that an IPE or an IPEEE
19 number or the sum of the two? Because the IPEEE numbers,
20 some of them, were quite large.
21 MR. THADANI: The intention was not only it's the
22 sume of IPE plus IPEEE, but it also should include shutdown.
23 That is, it is the overall core damage frequency, a mean
24 value. And that's what we had said, it would be a mean
25 value, which, by the use of the term, "mean," we imply some

1 knowledge of uncertainties in that. So the core damage
2 frequency of 10^{-4} include internal events, external events,
3 and low power and shutdown.
4 COMMISSIONER McGAFFIGAN: My recollection from
5 yesterday's briefing is that, on IPEEE's, we've reviewed
6 some, but we're still waiting for in the teens of
7 submittals, if I'm correct.
8 MR. THADANI: That's correct. That's correct.
9 COMMISSIONER McGAFFIGAN: So there's a large body
10 of folks who aren't at square one in this.
11 MR. THADANI: That's correct. There's a little
12 history to this. In June of 1990, there was an SRM which
13 gave some guidance. At that point, the Commission did not
14 want us to subdivide the so-called 10^{-4} in sub-allocating to
15 different contributors, so to speak.
16 Recognizing that there are significant questions
17 on methodology of shutdown risk, that that's not available,
18 that external events IPE's have not all been completed, so
19 what we have is a piece of the information.
20 One would have to sub-allocate -- make certain
21 arguments about what contribution there might be from
22 shutdown, from external events, recognizing what we have, by
23 and large, are IPE's that deal with internal events only.
24 This is again discussed in our guidance documents,
25 because that recognition is there, but the licensees would

1 have to pull together some arguments as to what those
2 contributions might be.
3 CHAIRMAN JACKSON: To bound them in some sense.
4 MR. THADANI: To some sense, yes.
5 CHAIRMAN JACKSON: A bounded contribution.
6 MR. THADANI: One needs to have nigh confidence
7 that those don't pop up as being the most significant
8 issues. We're going to be having, I expect, fairly
9 significant interaction with the Advisory Committee on these
10 issues next week. Our intention is to pull together some of
11 these thoughts and get the information to the Commission
12 soon after that.
13 CHAIRMAN JACKSON: Doesn't, to some extent, the
14 question revolves around what is the role of a numerical
15 analysis in a risk-informed, as opposed to a risk-based,
16 framework?
17 MR. THADANI: Yes. Yes. What's the role? We
18 made an attempt when we identified five basic principles in
19 our guidance documents that we would follow. That was an
20 integration of probabilistic approach, as well as
21 engineering analyses and what we call deterministic
22 thinking. How do we integrate that?
23 It seems to me that's a much better way to make
24 risk-informed decisions. It doesn't rely entirely on
25 numerical analysis aa the basis for decision, but that

1 numerical analyses do play a part in that decision.

2 CHAIRMAN JACKSON: There are two.

3 MR. THADANI: Unless there are some other general

4 questions, I think I'll just go to Tom King.

5 CHAIRMAN JACKSON: Why don't you go ahead.

6 MR. THADANI: Tom.

7 MR. KING: Thank you, Ashok. In fact, let me just

8 mention -- to follow up on your comment, Chairman Jackson --

9 some of the comments we did get from industry were related

10 to the role that we've put forward with using the PRA

11 numbers in a risk-informed fashion.

12 [Slide.]

13 Some of the comments were directed toward -- they

14 thought we've gone too far in using them in some sort of

15 decision criteria and that they really ought to be geared

16 toward looking at risk ranking, looking at trending and so

17 forth, and not hard and fast decision criteria.

18 CHAIRMAN JACKSON: Yeah, but then what do you do

19 if you talk about tech spec changes based on these analyses

20 or inservice inspection requirement changes based on these

21 analyses or inservice testing changes? So it sounds like

22 there's some variability in what the industry wants to do.

23 I mean, yeah, you can do kind of generalized risk-ranking

24 and sensitivity analyses and kind of configuration analysis.

25 That's one use. But if you're actually talking about

1 changes or relaxations in tech spec requirements or
2 inspection requirements, that's a different kind of thing.
3 MR. THADANI: Yes.
4 CHAIRMAN JACKSON: That's what I meant about what
5 is the use to which things are going to be put.
6 MR. THADANI: Yes.
7 CHAIRMAN JACKSON: And, therefore, that relates to
8 this question of quality, certainty, how much can be
9 tolerated, et cetera, et cetera.
10 MR. THADANI: If I may just add to just that
11 point, it is easier said than done. When I said the quality
12 should be driven by application -- which I think most of us
13 would --
14 CHAIRMAN JACKSON: Agree with.
15 MR. THADANI: -- agree is reasonable -- in some
16 areas -- for instance, inservice testing, quality assurance
17 -- that the issue propagates through the plant. It's not a
18 narrow issue, which means you have to rely on the overall
19 risk assessment. It is not just a small piece that we're
20 talking about. That means one has to have credibility in
21 the overall study, itself. And so, for that application,
22 the demand for the quality, it seems to me, would be very
23 significant.
24 On the other hand, if it has to do with an issue
25 -- let us just say tech spec changes on accumulators and

1 PWR's -- then I know that's a narrow issue. I know that
2 there are only a couple of sequences where that system plays
3 a part. I can make sure the quality of that analysis is
4 good. And, in fact, if the change in, say, core damage
5 frequency is very small, then one could have pretty good
6 confidence that the overall impact is, in fact, pretty
7 minimal.
8 ISD/QA issues of that type really, I think, go to
9 the heart of the broad issue of quality of the whole study,
10 not part of the study.
11 CHAIRMAN JACKSON: Right. I know Commissioner
12 McGaffigan has a question, but, actually, I'm going to put
13 my commissioner on the spot, because I know he has had some
14 fairly significant perspectives on this.
15 COMMISSIONER DIAZ: Oh, I do agree. I just really
16 would like to ask whether we are actually getting closer to
17 define whether there should be a level playing field on PRA
18 where, no matter what the application is, there is a base
19 quality that we can feel we can use risk information
20 consistently inside and outside, because I think that
21 becomes clearly an issue.
22 Until we can feel that we can use this across the
23 board with a certain level of quality and the industry
24 realizes that that will increase safety and it would also
25 reduce burden, we will always be writing things.

1 I think there has to be a demand that there be a
2 level playing field at some point, and we look for your
3 guidance in establishing what that level is. There has to
4 be. We can't be looking always at the minor application.
5 You have to have a base.
6 CHAIRMAN JACKSON: What's the based? Commissioner
7 McGaffigan.
8 COMMISSIONER McGAFFIGAN: It's really very closely
9 related, in that my sense is that the big dollar savings are
10 going to be in the complicated cases and where you're going
11 to need some sort of decent quality. They have to see that
12 that investment in having that quality PRA across the board
13 -- or at least for that plant, the IPE/IPEEE -- that that
14 will then result in savings.
15 If they don't see that, then probably they won't
16 go. But if they do see that, then maybe that up-front
17 investment can even be still made in a coming deregulated
18 environment. A good quality PRA -- could you remind me?
19 How much does a good quality PRA cost?
20 [Laughter.]
21 MR. THADANI: If you were starting from a clean
22 sheet -- which, in this case, we're not --
23 COMMISSIONER McGAFFIGAN: Right.
24 MR. THADANI: But if one were starting from a
25 clean sheet -- I would let Tom correct me if I'm wrong --

1 but the estimates I have got have been anywhere from 2 to
2 6 million dollars, depending on scope and so on, and if
3 we're not doing shutdown, that will save some pieces, as
4 well.
5 COMMISSIONER McGAFFIGAN: And how much does it
6 then cost to maintain that PRA, to keep it up to date -- in
7 an annual O&M cost? Do you know?
8 MR. KING: Well, just as an example, the South
9 Texas project people, they have a PRA group of around four
10 to five people that maintains it and supports risk-informed
11 applications and so forth.
12 COMMISSIONER McGAFFIGAN: So that would be about a
13 half million a year, maybe.
14 MR. KING: To get a ballpark idea.
15 COMMISSIONER McGAFFIGAN: Well, given some of the
16 applications that they want to apply this to, it strikes me
17 that that's a pretty good investment you could sell to a
18 board.
19 CHAIRMAN JACKSON: Right. It's a question of what
20 the saving is, if you could have a configuration. You don't
21 have to shut down some other things.
22 MR. THADANI: And we have heard some estimates on
23 different applications -- including, in South Texas,
24 technical specification changes -- as very significant
25 annual savings.

1 MR. KING: Right.

2 MR. THADANI: Very significant annual savings. So
3 an area where we're being told that the savings are not very
4 significant, I believe, was on inservice testing, actually.
5 That's what I was told by, I think, South Texas.

6 CHAIRMAN JACKSON: Yes.

7 MR. THADANI: But, in general --

8 CHAIRMAN JACKSON: But I do think that we have to
9 come out with some baseline and then, for the special big
10 applications, what additional is really required?

11 MR. THADANI: Yes. Yes. We have developed
12 guidance in terms of what our expectation was. Industry
13 doesn't agree. We have to deal with those issues next, the
14 significance of some of them.

15 CHAIRMAN JACKSON: All right, I'll adjourn the
16 meeting.

17 [Laughter.]

18 MR. KING: In conclusion, yes. Could I have
19 slide 2, please.

20 [Slide.]

21 What we're going to concentrate on today in the
22 briefing is what has happened over the past six months since
23 the last briefing in May. As you'll see, there has been a
24 lot of work on a lot of fronts.

25 We've made significant progress in the

1 risk-informed guidance documents. We've got some new
2 initiatives and activities we're going to talk about that
3 have been added to the quarterly update.
4 We've received several SRM's which we're going to
5 respond to specifically today in the briefing. I'm going to
6 focus on the major items, and then, at the end, we'll also
7 come back and talk about where we're going from here in the
8 future over the next few months. Slide 3.
9 [Slide.]
10 Since we briefed you last, you've received two
11 quarterly updates -- one in July and the one yesterday.
12 There has also been several papers and SRM's that have been
13 issued over that time frame.
14 Back in May, we had an SRM that asked the staff to
15 expedite activities on the use of IPE results in
16 prioritizing inspection activities, improving regional
17 capabilities, and providing inspector training. When we get
18 to slide 11, we'll address that specifically.
19 We had issued our draft regulatory guides for
20 comment and held a workshop in August. We'll talk more
21 about that later.
22 [Slide.]
23 Slide 4. On June 5th, there was an SRM that
24 requested our plans for training the NRC staff, which is a
25 very important activity to implement risk-informed

1 regulation. We've got several slides. We'll talk about
2 that activity, what has happened there and where we're
3 going.
4 On June 13th, we had an SRM that requested
5 progress reports on the voluntary industry effort to provide
6 reliability and availability data. We'll talk about that.
7 And, as I mentioned, we issued a couple of quarterly
8 updates. Slide 5.
9 [Slide.]
10 COMMISSIONER DIAZ: Let me just make a comment.
11 Let me put my professor's hat in here for a minute.
12 MR. KING: Yes.
13 COMMISSIONER DIAZ: And look at this, the training
14 and so forth. I haven't seen all of the final objectives on
15 it, but if I may think of how I used to deal with some of
16 these issues.
17 You know, I think one basic objective is, when you
18 finish this training, anybody in NRC that has to do with any
19 policy, decision-making, ruling, contact with licensee,
20 should have clear in his mind, when somebody says, "This is
21 risk-informed," that you have a picture.
22 When somebody says, "The coals are hot," or "Your
23 coffee is hot," you've got a picture. And when they say,
24 "risk-based," they should have a picture of what it is, and
25 when they say, "risk-informed, performance-based," it should

1 be clear.

2 MR. KING: Yes.

3 COMMISSIONER DIAZ: Because we keep mixing terms

4 or mixing terminologies and things. I think it has to be,

5 basically, that the NRC has to be, in this country, the

6 agency in which every technical person has a clear picture

7 of what each one of these things means and their

8 relationship to how we regulate.

9 CHAIRMAN JACKSON: Right.

10 COMMISSIONER DIAZ: Does that make sense?

11 MR. KING: I agree, I agree.

12 CHAIRMAN JACKSON: That's the basis. We can't do

13 anything otherwise.

14 MR. KING: Part of our training is directed toward

15 telling the staff, what do our documents say? What's

16 expected? How do we make decisions? Then part of it is

17 directed toward the technology of PRA's.

18 CHAIRMAN JACKSON: Right, but I think he's making

19 another statement, which I think is an important one, and

20 that is to get the definitions straight and that the

21 baseline of training people, before you get to the

22 technology use, is to have a common vocabulary that we all

23 understand.

24 If we're doing risk-informed, performance-based

25 regulation, PRA and the PRA Implementation Plan are tools

1 along that road, but they have to be put within the right
2 context, and so you need to be thinking about that, and
3 you're going to tell us about the training. But we have to
4 make an assumption, and you're going to hopefully give us
5 comfort in that assumption, that undergirding has to do with
6 clear definition.
7 MR. KING: If we can go to slide 5 and briefly
8 talk about the major accomplishments over the past six
9 months.
10 [Slide.]
11 Most recently, we completed the draft
12 risk-informed guide and SRP section on inservice inspection.
13 The Federal Register notice will be printed today announcing
14 the availability and announcing a workshop in November to
15 discuss these documents.
16 We've also made progress with some pilot
17 activities. We've received two applications so ar for
18 pilots, and I understand there may be up to three more
19 coming.
20 There have been a number of issues raised in the
21 Federal Register notice on inservice inspection that we're
22 soliciting comment on and will be discussed at the workshop.
23 These deal with issues like the scope of submittal,
24 degradation mechanisms and so forth, a number of technical
25 issues.

1 We received the Commission's October 1st staff
2 requirements memorandum, and we'll continue to work with the
3 industry on the two methodologies that are being developed
4 -- the qualitative and the quantitative -- and continue to
5 work with the pilot programs.
6 As I mentioned, we had held our public workshop in
7 August on the other reg guides and standard review plans,
8 and I'll talk more about those later.
9 We've completed work now on the IPE insights
10 report, what we call NUREG-1560. We had had a workshop in
11 April where we received 25 sets of comments. A number of
12 those required clarifications and expansion of the document,
13 and we've done that, and we're about ready to send it to the
14 printer's.
15 We included in our most recent quarterly update a
16 copy of the executive summary and the comments and responses
17 to the comments that were received. We've expanded our
18 training program -- and, again, we'll talk about that later
19 on. If I can go to slide 6, which now covers the new
20 activities that will show up for the first time in the
21 quarterly update you received yesterday.
22 [Slide.]
23 Over the past several months, we've had a couple
24 of meetings with NEI on an initiative that they've come in
25 with which involves using a full-scope PRA. By "full-scope"

1 I mean full power shutdown, external events, to look at
2 their plant, get a risk profile of the various systems an
3 components, compare it to the requirements, compare it to
4 their operations and maintenance cost, with the intent of
5 coming back and, in a risk-informed fashion, suggesting
6 changes to their current licensing basis.
7 What we've discussed with them are some of the
8 ground rules by which this study would be done, and they
9 also want to do three pilot projects to try out the review
10 and criteria which would be used to select and make
11 decisions on these on these items. We hope to finalize out
12 discussions and get this underway by December.
13 We've also had some discussions with the American
14 Society of Mechanical Engineering regarding development of a
15 national consensus standard for PRA that would cover scope
16 and quality. This would build upon the work we've done with
17 our draft NUREG-1602. We've worked with them and drafted up
18 sort of a charter for the group. The group involves not
19 just ASME people, but other people from industry, from
20 universities, as well as NRC.
21 COMMISSIONER DIAZ: Any key issues from trying to
22 bring the standards with 1602 together? Any great
23 differences, similarities?
24 MR. KING: Well, I'm not sure we're far enough
25 along. I could ask our person who attended the meetings

1 with ASME if she wants to expand on where we stand on that
2 activity. Mary. Mary Druin is from the Office of Research,
3 and she has been our representative working with ASME.
4 COMMISSIONER DIAZ: We're driving for consistency.
5 It would be nice to know how consistent we are.
6 MS. DRUIN: The ASME board met last Thursday and
7 did unanimously vote to develop a PRA standard. They
8 understand the criticality of the issue and our need to move
9 forward real swiftly in this area, so there is going to be
10 meetings, hopefully, like, biweekly. Biweekly -- is that
11 twice a month or is that twice a week? Anyway -- I always
12 get those terms mixed up -- twice a month.
13 CHAIRMAN JACKSON: Twice a week, if that is
14 biweekly. Bimonthly is twice a month.
15 MS. DRUIN: That's right. But hopefully it will
16 -- I mean we don't have the standard yet, but it is trying
17 to address the quality, the level of the detail, and get
18 into a lot of the issues that were discussed at the very
19 beginning of the meeting this morning.
20 CHAIRMAN JACKSON: Have they laid out a timeline
21 for their actions?
22 MS. DRUIN: The timeline that has been laid out is
23 to have a standard ready for NRC endorsement by December of
24 1998 -- in a year -- which, if we're successful, will be
25 phenomenal, because, typically, this is a four-year process.

1 They understand the criticality, and they're putting in a
2 new process to get this through in that kind of time frame.
3 CHAIRMAN JACKSON: And this is a better process
4 than doing it the reverse -- developing the standard
5 in-house and having a professional group review and endorse
6 them; is that right?
7 MS. DRUIN: I can't comment on that.
8 CHAIRMAN JACKSON: Mr. King?
9 COMMISSIONER DIAZ: But 1602 has the standards.
10 COMMISSIONER McGAFFIGAN: Implicit standards in
11 it.
12 MR. KING: Yes. And I think, from the comments
13 we've received, people feel that that's a good input to the
14 standards development effort, that it does have a good
15 foundation.
16 COMMISSIONER DIAZ: And one question was whether
17 there's going to be a significant difference between what
18 they are going to be doing, or do we have significant
19 similarities? In other words, are we on good grounds with
20 our standards or not?
21 MR. KING: I'm not sure we're far enough along to
22 say whether there's going to be a difference.
23 MS. DRUIN: Tom, we had a meeting last September,
24 and at the meeting were representatives from the different
25 owners' group, different utilities. NEI was present. One

1 of the things that was discussed was where do we start in
2 the standard?
3 Everyone was pretty much in a consensus to start
4 using 1602 work that has been developed by the CE Owners
5 Group and other organizations. There was not a lot of
6 diversity there, and no one felt that there was going to be
7 a big need to go out and create new writing, that there was
8 enough information out there, between all of these different
9 documents, that it was going to be more of a -- it don't
10 want to trivialize it -- but more of a cut-and-paste job.
11 CHAIRMAN JACKSON: Good. Thank you.
12 MR. KING: The third new activity, which is in
13 response to Direction Setting Issue 12 on risk-informed
14 regulation, is an effort by NMSS to develop a framework for
15 the application of risk-informed regulation. I understand a
16 paper will be coming to the Commission by the end of this
17 month providing that information.
18 CHAIRMAN JACKSON: Would you have any additional
19 comments?
20 MS. FEDERLINE: Yes. We would just let the
21 Commission know that we think the Commission's direction was
22 particularly timely in this regard. As you know, we had a
23 wide diversity of regulated systems in NMSS, all the way
24 from predictive 10,000-year analysis to the handling of
25 gauges.

1 So there are a couple of issues that we're going
2 to be bringing forward to the Commission that we've
3 considered. One is the diversity of the licensee base and
4 the economic motivation for turning to a risk-based
5 standard.
6 Another is the methodology questions. We're very
7 long in the development of waste disposal predictive
8 methodologies. We're less far along in the application of
9 human reliability in the medical applications, so there will
10 be a lot of development that needs to be done in the
11 methodology area. But we will be highlighting in this paper
12 these issues for you and proposing a path forward.
13 CHAIRMAN JACKSON: Good. Thank you.
14 MR. KING: The fourth items was, about a month
15 ago, we provided a paper to the Commission responding to
16 Chairman Jackson's July memorandum which followed up on an
17 ACRS letter that said we ought to consider elevating the CDF
18 to a level of a safety goal.
19 We've taken a look at that. We feel it's
20 certainly an item that might be very worthwhile in doing,
21 but in doing that there is a number of other issues that we
22 ought to look at in an integrated fashion, because the
23 safety goal policy talks about defense and depth; it talks
24 about uncertainties; a number of things that we're dealing
25 with now in trying to finalize these reg guides.

1 So we came to the Commission with a recommendation
2 to defer our answer on that until the end of March so that
3 we can deal with some of these related issues in the context
4 of finalizing the guides and then take those resolutions and
5 then come back with an integrated recommendation.
6 CHAIRMAN JACKSON: And that will be in the March
7 time frame?
8 MR. KING: End of March is what we proposed.
9 CHAIRMAN JACKSON: Right, because the current
10 safety goal policy, in fact, does not permit plant=specific
11 use.
12 MR. KING: That's correct. That's another issue.
13 CHAIRMAN JACKSON: And the Commission has recently
14 endorsed the plant-specific use.
15 MR. THADANI: Yes.
16 CHAIRMAN JACKSON: And that and this issue of
17 elevating the core damage frequency, which you're basically
18 de facto using on a plant-specific basis. You know, all of
19 these things have to be tied together, and I think you and I
20 have just said the same thing.
21 MR. KING: Yes.
22 [Laughter.]
23 And finally, we've started work now preparing for
24 a more intensive effort next year in looking at low power
25 and shutdown risk. This was prompted by an ACRS letter last

1 year. What we're doing this year is gathering existing
2 information from overseas, from industry, from other work
3 that has been done at NRC, and then, based upon that, well
4 decide what additional analysis we need to do.
5 CHAIRMAN JACKSON: Right. Now, I understand that
6 the staff, in fact, conducted fairly extensive evaluation of
7 low power and shutdown operations for both a PWR at the
8 Surry Plant and a BWR at Grand Gulf.
9 MR. KING: Correct.
10 CHAIRMAN JACKSON: And so, the question is, were
11 there inadequacies that were identified in those efforts
12 that would require us to embark on the new activity -- you
13 know, it's really related to what kind of scope of activity
14 are we planning for this risk study.
15 MR. KING: What we found out on that was two
16 things. We had done a screening study where we looked at a
17 number of plant states during shutdown and then tried to
18 identify the one that seemed to be most risky for the
19 detailed study.
20 In doing that, it became clear that the thing that
21 drives the risk is what the plant configuration is during
22 shutdown. It's not related to vendor type or containment
23 type or anything like that. It's how do they do the
24 refueling? The thing that we need to do --
25 CHAIRMAN JACKSON: In that was not a focus in the

1 earlier evaluations?

2 MR. KING: No. For the two plants we looked at,

3 we tried to identify that state, but it became evident that

4 that state does not apply to all the reactors out there.

5 CHAIRMAN JACKSON: I see.

6 MR. KING: So one of the things we need to do is

7 go back and see, are there some generic states that could

8 envelop the type of plants out there in the way they do

9 refueling and shutdown activities and try and take a look at

10 those conditions that we haven't looked at in these two

11 studies. That's one key aspect of what we need to do.

12 CHAIRMAN JACKSON: The focusing on this overall

13 plant configuration.

14 MR. KING: Yes.

15 CHAIRMAN JACKSON: Commissioner Diaz.

16 COMMISSIONER DIAZ: Yes. And, of course, when we

17 do this, we're going to set a standard for the industry.

18 We're going to do it in a very consistent, realistic,

19 thorough basis where apples are apples and oranges are

20 oranges, because it's not,

21 MR. KING: Correct.

22 COMMISSIONER DIAZ: Thank you.

23 CHAIRMAN JACKSON: And that that's the point of

24 trying to genericize some of this.

25 MR. KING: That's the point of trying to

1 genericize, but genericize in a sense that really reflects
2 the way things area done in plants. I don't know whether
3 we'll end up with six types of conditions or four types, but
4 that remains to be seen at this point.
5 CHAIRMAN JACKSON: Okay.
6 MR. THADANI: I would just make a comment,
7 Commissioner Diaz, that yes, indeed, but, in terms of
8 uncertainties, that becomes, I think, a much more
9 significant issue during shutdown of low power operation
10 because they're largely controlled by, A, human actions; B,
11 some of the models that need some work, like improvements in
12 trying to estimate these risks.
13 MR. KING: Let me go on to slide 7.
14 COMMISSIONER DIAZ: I am not going to respond
15 because it will consume the rest of the meeting.
16 [Laughter.]
17 CHAIRMAN JACKSON: Well, but before you do go on,
18 I guess I'm interested in implementing the Commission's
19 policy statement. Has the staff envisioned any
20 risk-informed applications that would require changes to the
21 backfit rule? If you haven't thought about it --
22 MR. THADANI: The way we have gone forward has
23 been that this is a voluntary option for the industry, and
24 that's how the guides are structured. They're not
25 impositions, but that if the industry wants relaxations,

1 this is an approach we would pursue. We have not thought
2 through the backfit implications if we were to impose such
3 an approach as being mandatory. Currently, we're pursuing
4 just the voluntary option.
5 CHAIRMAN JACKSON: Well, maybe you need to do some
6 background on that.
7 MR. THADANI: We'll give some thought to that
8 issue.
9 CHAIRMAN JACKSON: Right.
10 MR. KING: One thing we have thought about -- at
11 least the team working on these guides -- was the regulatory
12 analysis guidelines.
13 CHAIRMAN JACKSON: Right.
14 MR. KING: Maybe they need to be updated once we
15 get the framework and the principles and guidelines in
16 place. For example, they don't really talk about defense
17 and depth.
18 COMMISSIONER McGAFFIGAN: It's fascinating.
19 MR. KING: We have had some discussions internally
20 on that.
21 MR. THADANI: I want to make sure I didn't
22 misunderstand your question. Your question was to the heart
23 of the backfit rule, itself.
24 CHAIRMAN JACKSON: Yes.
25 MR. THADANI: To the heart of the backfit rule.

1 That's what I thought. Thank you.

2 CHAIRMAN JACKSON: You answered the question.

3 Thank you.

4 MR. KING: If I could have slide 7, please.

5 [Slide.]

6 Slide 7 talks about the guides and standard review

7 plans that are out for comment now. That comment period

8 closed a couple of weeks ago. We now have about 30 sets of

9 comments, as Ashok mentioned. Generally, the comments are

10 supportive of the concept, the approach, the structure of

11 the documents.

12 Now, they did have a lot of detailed comments on

13 specifics of the decision criteria and things that were not

14 clear to them, and I think we probably will be making a

15 number of changes, certainly, to clarify things like, we got

16 a lot of questions on the use of NUREG-1602, a lot of

17 misunderstanding, that kind of thing.

18 CHAIRMAN JACKSON: Let me ask you a question. In

19 terms of going back to the issue we talked about a little

20 bit earlier, I want to be sure I understand some of the

21 concerns with respect to required PRA quality. Was the

22 concern having to do with the fact that the required quality

23 was not adequately described, or was the concern with the

24 quality as laid out already in the documents?

25 MR. KING: The concern was that what we asking for

1 was the gold-plated PRA, that our putting forth NUREG-1602
2 was interpreted as requiring a gold-plated PRA. That was
3 not our intent.
4 CHAIRMAN JACKSON: Right.
5 MR. KING: And we need to clarify that.
6 CHAIRMAN JACKSON: And then, this issue of the
7 acceptance guidelines for very small or negligible risk
8 increases, is that an issue for those plants that may be
9 bumping up against the 10^{-4}, as opposed to -- because it
10 strikes me that you've dealt with it, essentially, in the
11 guidance, as far as I understand it, but there is an issue
12 for those plants that either today are at or maybe even
13 slightly exceed the 10^{-4} core damage frequency.
14 MR. KING: It's an issue for all plants that want
15 to use risk-informed regulation, but I think it's of
16 particular importance to those that are bumping up against
17 the 10^{-4}.
18 CHAIRMAN JACKSON: And in looking at that, have
19 you looked at issues that may have to do with -- let's call
20 it for those there for the moment, for the purposes of a
21 straw man -- looking at risk neutrality, in terms of where,
22 if there were some risk increase in one place, there might
23 be some compensation somewhere else so that that is a
24 potential way, within some band, of handling things without
25 losing something that you tag your analysis to?

1 MR. KING: Yes. In fact, that's one area we need
2 to clarify in our documents. We agree with the concept of
3 bundling several changes together -- some risk increases,
4 some risk decreases. We think that, certainly, to do that,
5 that will bring risk down in some areas, which is one of the
6 incentives for allowing bundling.
7 CHAIRMAN JACKSON: Right, but it also allows you
8 do deal with the issue in a more generic way, but
9 particularly for those that are bumping up against some
10 ceiling.
11 MR. KING: Yes.
12 MR. THADANI: Commissioner.
13 CHAIRMAN JACKSON: Go ahead.
14 MR. THADANI: I just want to make sure -- and I
15 believe that is completely consistent with the policy
16 statement.
17 CHAIRMAN JACKSON: Right.
18 MR. THADANI: Because the policy statement said
19 you need to focus on both -- the areas where we need to
20 relax our requirements and the areas it may be appropriate
21 to enhance safety.
22 CHAIRMAN JACKSON: Right. In fact, the lawyers
23 would tell you that you can't go one way without going the
24 other.
25 MR. THADANI: Yes, I've seen this.

1 COMMISSIONER McGAFFIGAN: How much does a
2 gold-plated PRA cost?
3 [Laughter.]
4 MR. KING: That's probably Ashok's $6 million.
5 COMMISSIONER McGAFFIGAN: Is the $6 million one --
6 CHAIRMAN JACKSON: I don't know that you -- can
7 you really make such a statement? Because it's a plant-
8 specific issue.
9 MR. THADANI: I think it's very difficult for us
10 to sit here and give you that estimate, because their IPE's
11 have already been done, and there is variability, as I think
12 the Chairman noted, in terms of the quality, so the cost of
13 upgrading -- we don't know, for that matter, the quality of
14 the IPEEE's.
15 COMMISSIONER McGAFFIGAN: Right.
16 MR. THADANI: And so the cost of upgrading could
17 be from a small amount to possibly quite significant. I
18 think it's very hard to give you one --
19 CHAIRMAN JACKSON: Well, is the definition of
20 gold-plated that I've done my PRA --
21 COMMISSIONER McGAFFIGAN: And it's good enough.
22 CHAIRMAN JACKSON: And it's good enough, and I
23 don't want to change it? And if you ask me to change it for
24 some application, by definition that makes it gold-plated?
25 MR. THADANI: No. I think industry's roles are

1 gold-plated, but --

2 CHAIRMAN JACKSON: No, I'm not talking about your

3 point of view. I'm playing the devil's advocate here a

4 little bit, not with you, but in the more generic sense.

5 MR. KING: I don't think changing it to reflect or

6 support an application is an issue. I think it's what's

7 that base level of quality.

8 CHAIRMAN JACKSON: It's the base level issue

9 again.

10 MR. KING: Yes.

11 COMMISSIONER DIAZ: But definitely we want it to

12 be corrosion of the system; right?

13 [Laughter.]

14 MR. KING: True.

15 COMMISSIONER McGAFFIGAN: Yes.

16 CHAIRMAN JACKSON: Gold is good in that regard.

17 Brass, on the other hand, does tarnish.

18 MR. KING: The comments we receive, we'll be

19 discussing those with ACRS next. We'll also, when we come

20 back with the proposed final documents, be summarizing them

21 for the Commission, as well, and what our response is.

22 Let me talk a little bit about the last bullet,

23 the policy issues. There are several items that have come

24 out of the comment process and from internal discussions

25 that we're going to come back to the Commission with to get

1 a policy decision. We've got a paper under preparation now.
2 We hope to have it up here by the end of this month.
3 Two of the issues are listed here -- treatment of
4 uncertainties, which we talked about earlier, and acceptance
5 guidelines for very small or negligible risk increases.
6 That gets, really, to the question of what's the
7 definition of risk-neutral. Can very small increases in
8 risk essentially, from a practical standpoint, be considered
9 risk-neutral, which would allow more flexibility for plants
10 that are bumping up against the 10^{-4} CDF, for example, to
11 come in and participate in risk-informed changes.
12 CHAIRMAN JACKSON: What is the complaint about the
13 treatment of uncertainties? You haven't treated them
14 anyway, so -- at this stage of the game.
15 [Laughter.]
16 So what's the complaint or the potential
17 complaint?
18 MR. KING: The complaint is that we've asked for
19 too extensive an uncertainty analysis in the current draft.
20 CHAIRMAN JACKSON: In the current draft?
21 MR. KING: Yes.
22 CHAIRMAN JACKSON: Okay.
23 MR. KING: What we're thinking about at this
24 point, and which we'll talk about in this policy paper is
25 very small changes in risk increases, very small increases,

1 how extensive an uncertainty analysis do we need? Can we
2 define a small range that we can call risk-neutral where the
3 uncertainties really don't matter, because the change is so
4 small, and, therefore, you don't need as extensive a
5 treatment of uncertainties.
6 It would also allow plants that are bumping up
7 against the 10^{-4} to come in and propose changes.
8 CHAIRMAN JACKSON: But isn't it rooted in the fact
9 -- and we're not going to sit here and debate it all day or
10 anything -- but isn't it related to the confidence you have
11 in the beginning? With the answer, I mean.
12 It is naive to think that you can just take a mean
13 and say, "Okay. That's it. I don't know how well I know
14 that mean, and it's okay." So it's small, and it may be,
15 though, that there's a band around it where you are that's a
16 factor of 10 higher. And so it is not an issue that you can
17 sweep under the rug.
18 MR. KING: No. And part of the problem is those
19 same concerns apply when you're talking a confidence level,
20 because even with a full-scope PRA that includes external
21 events and shutdown, there are things that are not analyzed,
22 that are unanalyzable at this point -- management and
23 organization factors, aging of components, probably some
24 others.
25 So when you're talking mean or you're talking a

1 confidence level, you still have to somehow decide how
2 you're going to treat those unquantifiable factors. So
3 we're trying to come up with a scheme that deals with that
4 and that also deals with the application that makes sense
5 from the standpoint of maybe they're proposing a change that
6 really is only affecting full power operation, and they
7 don't want to do a low power and shutdown. How do you deal
8 with that?
9 CHAIRMAN JACKSON: Okay. I understand.
10 MR. KING: It's a complicated problem.
11 CHAIRMAN JACKSON: Right. Go ahead.
12 MR. THADANI: I might note, in our paper that we
13 sent you just a few days ago, there is an attachment that
14 talks to this issue and some of the initial thoughts, at
15 least, on how we might want to consider proceeding.
16 MR. KING: Yes.
17 CHAIRMAN JACKSON: When will the Commissioners
18 receive your formal analysis of the comments and your
19 proposed resolution -- and pulling out the policy issues?
20 MR. KING: What we were doing is pulling out the
21 policy ones and sending them up the end of this month.
22 CHAIRMAN JACKSON: Okay.
23 MR. KING: That's our plan. The others, both the
24 comment and how we've treated it, we're planning to put
25 forward in the package that sends the final documents to the

1 Commission, which is December.

2 CHAIRMAN JACKSON: Right. All right. So this is

3 all tracking, still, to have the finalization by the end of

4 the year.

5 MR. KING: Yes.

6 CHAIRMAN JACKSON: Commissioner McGaffigan.

7 COMMISSIONER McGAFFIGAN: The second policy issue,

8 the acceptance guidelines for very small or negligible, does

9 this get into things like 10^{-6} or -- I mean have you chosen

10 a number for what is very small or negligible that you've

11 quantified and said, "Okay. If it's 10^{-6}, I don't care

12 whether they're bumping up against 10^{-4}, because 10^{-6} is

13 1/100 of 10^{-4}, so therefore I'll consider that negligible in

14 the scheme of things"?

15 MR. KING: We need to define a number.

16 COMMISSIONER McGAFFIGAN: You need to define a

17 number? You do not have a number?

18 MR. KING: I've chosen a number, but that doesn't

19 mean that we have a consensus on it.

20 COMMISSIONER McGAFFIGAN: Is it fair to ask what

21 that number is?

22 [Laughter.]

23 CHAIRMAN JACKSON: No, because it's his number,

24 not their number.

25 MR. THADANI: Yes, right. And I want to be

1 careful. I think that's not the only variable. The other
2 variable is how many times and the cumulative impacts --
3 CHAIRMAN JACKSON: And what's the cumulative
4 impact?
5 COMMISSIONER McGAFFIGAN: Right.
6 MR. THADANI: And that's the real issue, I think,
7 that we have to pay attention to.
8 CHAIRMAN JACKSON: Exactly.
9 MR. KING: And there may be -- we've listed two
10 policy issues.
11 CHAIRMAN JACKSON: I mean risk neutrality is
12 defined within that context, anyway.
13 MR. KING: Yes.
14 MR. THADANI: Yes.
15 CHAIRMAN JACKSON: Yes.
16 MR. KING: This would define risk neutrality,
17 whatever that number is.
18 CHAIRMAN JACKSON: Exactly. That's right.
19 MR. KING: And there may be other policy issues in
20 the paper, as well. We're talking about, for example, do we
21 need separate guidelines for the shutdown condition? Do we
22 need guidelines to cover temporary increases in risk? So
23 the paper in October will deal with all of those.
24 [Slide.]
25 Let me go on to slide 8 and just quickly talk

1 about where we stand on IPE and IPEEE. The IPE reviews are
2 done except for Browns Ferry Unit 3, which will be done in
3 December, I believe.
4 We are putting together what we call the IPE
5 follow-up program, which is talking a look at a number of
6 the results from the standpoint of the plants that had a
7 relatively high core damage frequency or containment failure
8 probability.
9 Perhaps we would want to follow up with those and
10 see why haven't they done something to reduce that using the
11 guidelines in the regulatory analysis guidelines document as
12 sort of a benchmark to look at what improvements could be
13 made.
14 There are some generic issues that perhaps we want
15 to work on. You know, probably the most prominent one is
16 the pump seal LOCA for PWR's. A number of plants, that was
17 a dominant sequence. Do we want to do something further
18 there.
19 We had asked plants to specifically answer some
20 what we called containment performance improvement
21 questions, questions that came out of generic studies that
22 were done several years ago.
23 A number of plants answered those; a number of
24 plants didn't. We want to follow up on the ones that
25 didn't, and we want to follow up and see have licensees

1 really done the things that they committed to do when they
2 submitted their IPE, make the improvements they said they
3 were going to make.
4 So those are the kinds of things that we plant to
5 include in our follow-up program, and we owe you a separate
6 paper, giving more detail on that. I think it's in
7 November.
8 The IPEEE is underway. We're going to give you an
9 interim insights report in November. We've just tried to
10 put a short executive summary in the paper that you received
11 yesterday, and ultimately, when we're done, we'll have a
12 follow-up program similar to the one we're embarking on on
13 the internal events.
14 CHAIRMAN JACKSON: Let me ask this quick question.
15 Are the IPE results becoming obsolete, and do we know how
16 many licensees have been updating their IPE's? Because kind
17 of undergirding this is the issue of, if we are going to
18 make regulatory decisions based on PRA results -- which is
19 what most of these IPE's have turned out to be -- there's an
20 issue there in terms of -- and you've mentioned cumulative
21 impact, for instance, of changes. Are we thinking about how
22 that's going to be tracked and what that implies about how
23 updated the IPE's need to be?
24 You know, I've been to nuclear plants this year.
25 This is 1997. Now, admittedly, what may be in our

1 residents' offices may be different than what's on the shelf
2 in the licensees' engineering organization or PRA
3 organization, but I've seen dates like 1991. Presumably,
4 there have been changes made to the plant, including ones
5 that could have lowered the estimated core damage frequency,
6 as well as ones where there has been no analysis one way or
7 the other to know what the cumulative impact is. What are
8 we doing about that?
9 MR. KING: The IPE's, basically, are a snapshot or
10 information that's maybe five years old, something of that
11 nature.
12 CHAIRMAN JACKSON: Well, I guess I'm really asking
13 something else, which has to do with, if we're purporting to
14 make regulatory use of the PRA's -- and I want to stay on
15 that plane, but I'll ask it within the context of the IPE's
16 -- is there not a linked question having to do with living
17 PRA's, how they're maintained and how up-to-date they need
18 to be relative to -- otherwise, how do you make the
19 judgment?
20 MR. KING: Our guidance documents --
21 COMMISSIONER DIAZ: Going back to what is not even
22 a dead horse anymore -- it's a skeleton by now.
23 [Laughter.]
24 But it goes back to establishing a base --
25 CHAIRMAN JACKSON: Baseline, yes.

1 COMMISSIONER DIAZ: Okay. And they need to know
2 what the baseline is, and we need to inform them of what it
3 is so we can say this is what we will find a baseline
4 acceptable to make these changes.
5 CHAIRMAN JACKSON: Right. And included in that is
6 the issue of how do you update?
7 COMMISSIONER DIAZ: Yes.
8 CHAIRMAN JACKSON: I mean we're dealing now -- I
9 mean that's our big lesson learned in the last year and a
10 half with updating and maintaining certain fundamental
11 information vis-a-vis the licensing basis and design basis
12 information. The question is, what are we doing?
13 MR. KING: There is no requirement for them.
14 CHAIRMAN JACKSON: No, no, no. I'm not talking
15 about the requirement.
16 MR. KING: I mean there was no requirement for
17 them to do a specific type of PRA in response to an issue
18 generically in the first place.
19 CHAIRMAN JACKSON: Well, no. That is not the
20 issue. One is an informational question. One, are the
21 IPE's becoming obsolete? The second part of the
22 informational question, do we even know how many licensees
23 are updating their IPE's -- second informational question.
24 Because, three, it has bearing on the third
25 question, which is, if you're going to lay out standards

1 having to do with what the IPE's have to be for what
2 regulatory use you're going to make of them -- because
3 that's what we're talking about -- you have to include in
4 there how they're updating.
5 I'm not saying you're laying it as a requirement,
6 but you have to have something. You can't take 10-year-old
7 information and make a regulatory judgment on it. That's
8 all I'm saying.
9 MR. THADANI: I think we can answer one out of
10 three questions.
11 [Laughter.]
12 CHAIRMAN JACKSON: Well, that's good.
13 MR. THADANI: I'm pretty comfortable with the
14 second question you have raised, which is, first of all, are
15 there some licensees whose IPE's or obsolete or are there
16 licensees who are keeping up? I think the answer probably
17 is yes to both, but how many I don't think we could say.
18 Clearly, the pilot plants that we've been working with have
19 been keeping up and have been paying attention to the issue
20 of quality and so on.
21 My sense would be that if we keep moving in this
22 direction where there is, in fact, consensus within the
23 industry, more and more of the licensees' IPE's would be
24 along the track.
25 The only other comment I wanted to make was, in

1 the guides that we have, we have clearly stated that when
2 licensees are coming in for changes that the analysis,
3 itself -- that is the risk assessment -- should reflect the
4 plant as is and not as it might have been.
5 CHAIRMAN JACKSON: So, in a sense, you have that
6 covered in there.
7 MR. KING: It's in there now, yes. Yes.
8 CHAIRMAN JACKSON: That's the baseline issue. Mr.
9 Callan, you were going to make a comment.
10 MR. CALLAN: I was just going to say -- and I'm
11 sure Ashok would agree -- that we can't take comfort from
12 the fact that pilot sites are maintaining their IPE's,
13 because they were selected precisely because they maintained
14 their IPE's current.
15 CHAIRMAN JACKSON: Yes.
16 MR. CALLAN: That they are, in effect, leaders in
17 the industry in that regard. So I don't think that's a good
18 indicator of the rest of the industry.
19 CHAIRMAN JACKSON: You said something very
20 important, though. You've already said, in the guidance
21 documents that you've developed, that you already clearly
22 have stated that when one comes in with a PRA it has to
23 reflect the latest and the greatest.
24 MR. CALLAN: As built, as operated plant is what
25 it says.

1 CHAIRMAN JACKSON: Right. In order to be able to
2 use it. Commissioner McGaffigan.
3 COMMISSIONER McGAFFIGAN: My only point is that
4 the standard process -- the ASME standards process that we
5 talked about earlier that's going to start with 1602 -- if
6 you have imbedded in 1602 that a good PRA is a living PRA,
7 then that presumably is the standard we're going to be
8 propagating into standard space.
9 I think that's good, but I suspect some of the
10 stuff we're reading in Inside NRC about gold-plated PRA's
11 may well be -- a living PRA is -- I'm trying to still get a
12 definition of gold-plated.
13 [Laughter.]
14 Maybe part of it is that it's living and it has to
15 have been updated since the last time you thought about
16 this. Many of them see that large improvement as too hard,
17 as opposed to the South Texases who you've been working with
18 who, for them it's a small increment, and they are ready to
19 go. But I think it's real important, the notion of a living
20 PRA.
21 MR. THADANI: Yes. And, in fact, what Tom said
22 earlier is significant. South Texas, I think having a group
23 of five or six people, it is that the idea is not just to
24 update the PRA; the idea is to apply it.
25 COMMISSIONER McGAFFIGAN: Right.

1 CHAIRMAN JACKSON: Right.

2 MR. THADANI: And so if a licensee were to apply,
3 there is obviously the incentive to make sure it's kept up,
4 it's in fact a living PRA.

5 CHAIRMAN JACKSON: Right.

6 MR. THADANI: The issue is going to be --

7 CHAIRMAN JACKSON: Okay. You know, I've asked on
8 several occasions, but I'm now going to get you to commit to
9 this. Are we keeping record of what regulatory use we are
10 have been making of IPE results? I've asked this question
11 for -- two years. So now I'm going to get you to commit to
12 a date. Or we'll give you a date.

13 MR. KING: No. We are committed. You will find
14 it in the table in the quarterly update.

15 CHAIRMAN JACKSON: Ah, okay. Very good. We've
16 been talking about South Texas, and I notice it's on the
17 next viewgraphs.

18 MR. KING: Yes. I'm going to turn it over to
19 Scott Newberry, who will talk about the pilots.

20 MR. NEWBERRY: Yes. Good morning.

21 CHAIRMAN JACKSON: Good morning.

22 MR. NEWBERRY: I'm going to go through the status
23 of the four pilots on the next couple of viewgraphs and then
24 a little bit about insights in the inspection program.

25 Of course, each of the pilots would use a

1 risk-informed approach to all their NRC requirements and the
2 associated program at the plant in response to those
3 requirements, using a process along the lines of the Reg
4 Guides which are out for public comment, and I know there
5 are issues, and are being finalized, which creates a
6 challenge in the pilot process. But progress has been made.
7 The first pilot, on tech specs, just to remind you
8 that that pilot would extend the outage time for certain
9 equipment, ECCS equipment that would be used to respond to
10 unlikely events, large LOCA's or the safety injection
11 accumulators and low pressure injection systems.
12 The Commission -- I guess early last summer --
13 approved the Arkansas risk-informed tech spec SER. But
14 there was an issue in the SER that we're working with
15 licensees on that has to do with one of the five principles
16 in a risk-informed approach, and that's the configuration
17 risk management program that would be committed to by the
18 licensee.
19 I think the snapshot of our experience to date is
20 that that's really a plant-specific issue. In working with
21 specific licensees on that, it looks like -- San Onofrio, I
22 think, may become the lead plant there -- that we should be
23 able to finalize a position on that issue and get our first
24 safety evaluation out by the end of the year that would
25 approve a plant-specific configuration risk management

1 program that would serve as a model -- perhaps a simple
2 example here, but a model nonetheless -- that shows that we
3 can make a risk-informed decision with tech specs.
4 Graded quality assurance pilot, we just sent a
5 paper up to the Commission -- 97-222 -- which forwards the
6 draft safety evaluation report for South Texas. This SER
7 would accept Houston Power and Light's risk-informed
8 revision to their operations quality assurance program.
9 This was an interesting pilot. I think it
10 challenged the staff to use the guidance, to learn from the
11 guidance. Many of these issues that came up here were
12 talked about extensively as we moved that safety evaluation
13 report up to the Commission.
14 I think the conclusion in the safety evaluation
15 report is important to point out to you, from the standpoint
16 of including, that we think this program presented an
17 overall safety improvement at the plant.
18 But the issues of -- I think the term, "bundling,"
19 was used here -- clearly, that activity was meant to focus
20 on the most important equipment at the plant so there's
21 intense activity, but to relax the program on the less
22 important equipment at the plant.
23 But then there's an increased feedback mechanism.
24 -- that's principle number 5 in NUREG-1060 -- which we think
25 will really provide us a safety benefit.

1 I think those conclusions really come from the
2 judgment of the staff. I think there's a lot of qualitative
3 evaluation involved in this SER. So we were happy to get
4 that SER to you finally in that recent paper. Next
5 viewgraph please.
6 [Slide.]
7 IST and ISI pilot. Both pilots, of course, would
8 be intended to improve test programs, inspection programs at
9 plants using risk insights. We're still shooting for a
10 December date on the first IST pump and valve pilot at
11 Comanche Peak.
12 In terms of recent activities, we had a team on
13 site at Comanche Peak dealing with the issue of PRA quality.
14 Looking at the PRA for this particular application, there
15 were some minor issues identified that are being worked
16 through that have to do with elements of the PRA such as
17 treatment of human performance, success criteria, issues
18 like that.
19 I think I would mention that the dates have
20 delayed a little bit on this pilot and others. Utilities
21 were focusing resources on providing comments to us on the
22 Reg Guides and SRP's so that there has been an impact there.
23 But, as I said, we're still shooting for December on that
24 pilot for Comanche Peak. Palo Verde is going to slip into
25 1998.

1 Inservice inspection of piping. I want to modify,
2 I think, the context of that first bullet there, in terms of
3 "nothing received for review yet." At the time of the
4 viewgraph, nothing had been received for the identified
5 pilots to date. Those would be Surry, ANO-2, and
6 Fitzpatrick.
7 However, just within the last couple of days,
8 ANO-2 did come in with a full proposal, and we understand,
9 in talking to Surry, that they would be in next week. And
10 then, a recent letter from the industry adding two more
11 pilots, ANO-1 and Vermont Yankee, so there would be five
12 plants pursuing ISI initiatives.
13 Now that we have ANO-2 and with Surry coming in,
14 we'll be able to look at schedules and priorities and
15 provide you an update in the next plan. Next viewgraph,
16 please.
17 [Slide.]
18 We've got one viewgraph here on some of our
19 actions in response to your May SRM regarding use of PRA in
20 the inspection program. I think it's fair to say, too --
21 and, Chairman Jackson, you mentioned that -- we're really
22 talking about a philosophy here in terms of implementing the
23 policy statement.
24 So there are other broader actions in our pursuit
25 of opportunities every day, not just in the inspection

1 program, but, of course, here at headquarters, to try to use
2 risk-informed decision making in our programs, but you hear
3 it particularly in response to your SRM.
4 You asked about prioritizing inspection activities
5 using IPE information, improving the region's ability to use
6 risk insights, and then, of course, staff training. Pat
7 Baranowski will talk in some detail about training after I
8 finish up here in a minute.
9 But in terms of particular staff actions to date,
10 a lot of the information that has been talked about here in
11 terms of the IPE's, the information that we have has been
12 made available now through documents. There will be more
13 documents going out to the regions, but there have been
14 training lectures at all the regions conducted by people
15 familiar with the IPE results.
16 Then there's a continuing effort with the senior
17 reactor analysts program, keeping that program staffed.
18 CHAIRMAN JACKSON: No, do all the regions, at this
19 point, have senior reactor analysts?
20 MR. NEWBERRY: Yes. The exact status on that is,
21 of course, there's 10 SRA positions in the agency -- two at
22 headquarters and, so, two at each region.
23 CHAIRMAN JACKSON: So there's at least a body at
24 each region.
25 MR. NEWBERRY: Yes. Region 3 has two SRA's, but

1 they're not certified yet. They're in the training program.
2 CHAIRMAN JACKSON: I see.
3 MR. NEWBERRY: That's an 18-month program.
4 CHAIRMAN JACKSON: And to whom are the training
5 lectures targeted?
6 MR. NEWBERRY: As I recall, it's a broad target.
7 We look for opportunities to bring people in from the field,
8 residents as well as regional staff. So the effort here --
9 and you'll see that, I think, in the training.
10 There's training directed toward the review staff
11 and NRR in terms of what these guidance documents say.
12 There's training on PRA technology. There's training for
13 inspectors. There's training for managers. There's a broad
14 spectrum of training, and Pat's going to talk about that.
15 The intent here, and I think you'll see it in Pat's
16 viewgraphs, is to touch on, basically every NRC employee
17 making regulatory decisions here.
18 CHAIRMAN JACKSON: Okay.
19 MR. NEWBERRY: Guidance on the use of PRA in the
20 Inspection Manual. Just last month, an appendix to Manual
21 Chapter 2515 was completed. Just looking at it recently, I
22 think folks will find that very interesting. We will
23 probably get some feedback on it and maybe have to do
24 something else.
25 But there's everything from general advice to

1 advice to a glossary of terms in that Manual chapter. There
2 are examples of things that have been found and could be
3 found at plants with respect to support system
4 relationships, to front line systems, and also guidance on
5 ranking issues in planning using PRA information. And, as I
6 said, Pat will be talking about training here in a minute.
7 Upcoming actions -- the training effort is a
8 pretty significant investment of resources, both in the
9 training center facilities and also getting employees into
10 those classes and the seminars. It's an important activity.
11 We're just starting to do more in revising core
12 inspection procedures. In particular, the graded QA
13 inspection procedure is being worked on, and that's
14 explicitly an item in the plan, and that's due to be
15 completed this spring.
16 CHAIRMAN JACKSON: Good. Let me ask you a quick
17 question. You talked about the ANO risk-informed tech spec
18 changes.
19 MR. NEWBERRY: Right.
20 CHAIRMAN JACKSON: How long did that take?
21 MR. NEWBERRY: I don't know. Is there somebody
22 here?
23 CHAIRMAN JACKSON: And are we --
24 MR. NEWBERRY: We can get you an answer on that.
25 CHAIRMAN JACKSON: And even though the guidance

1 documents that you're working on are still being finalized,
2 the question is are you gathering kind of what you've
3 learned so that you kind of begin to develop a better oiled
4 process for doing them to shorten the time?
5 MR. NEWBERRY: Yes. In fact, some of the comments
6 received have been from staff working on those reviews.
7 CHAIRMAN JACKSON: Okay. And also input from
8 licensees to try to draw it all together into a process.
9 MR. NEWBERRY: Yes,. Some of the comments received
10 were derived directly from the pilots -- the tech spec
11 pilot, for example.
12 CHAIRMAN JACKSON: So if I talked to ANO and so
13 on, what would they say? Were they pleased as Punch or --
14 you know, "This is the cat's pajamas"?
15 MR. NEWBERRY: No. I will take a chance here and
16 guess at what they would say. I would say they would be
17 concerned about the risk configuration management program,
18 that we should limit ourselves to, perhaps, the extension.
19 More precisely, I think, for example, the safety
20 injection accumulators now have a one-hour AEOT; we would
21 extend it to 24 hours. They would like to limit that safety
22 assessment or that configuration issue to the latter part of
23 that extension. So they would have an issue. They would
24 have an issue.
25 CHAIRMAN JACKSON: Okay. Training. Pat.

1 [Slide.]

2 MR. BARANOWSKI: The October 14th update to the

3 PRA Implementation Plan briefing provided an attachment that

4 talked about training, and I have a few highlights here.

5 That attachment responds to the June 5th SRM in

6 which we were requested to discuss plans for training the

7 staff and, in particular, with regard to training for

8 regulatory approaches that would be relevant to the

9 risk-informed regulatory guides and the standard review

10 plans that are in development, as well as overall training

11 for basics in PRA, and, moreover, to focus somewhat on

12 regional inspection training activities.

13 The PRA Implementation Plan does include several

14 tasks related to training. These are modified as we go

15 along to reflect the regulatory program that's in

16 development as it evolves.

17 We have made some changes over the last several

18 months and are continuing to define and implement some new

19 training requirements, and I have a couple of them that I'll

20 mention here, in particular, with regard to the NRR

21 technical staff.

22 First, let me mention that there is a seminar that

23 has been put together that covers the responsibilities

24 associated with risk-informed regulatory activities, and

25 it's designed to familiarize the NRR staff in general, and

1 hopefully it will cover the kinds of things that we talked
2 about a little bit earlier -- terms and things that we can
3 all talk in a common language.
4 It's a mandatory seminar. It's meant to motivate
5 and familiarize the staff with the uses of risk-informed
6 regulatory initiatives, and it's primarily taught by an NRR
7 senior manager. I think Gary Holohan has done the most
8 recent one.
9 MR. NEWBERRY: Let me just make a comment quickly
10 there, Pat, in response to Commissioner Diaz. Your issue on
11 this mental image of risk-informed comes out very quickly in
12 the dialogue that's created in that seminar. That seminar
13 is built around regulatory policy, the PRA policy statement,
14 and the intent of where we're trying to go.
15 We've got a ways to go, based on the dialogue in
16 those seminars, but I think that's where it's beginning to
17 take place.
18 CHAIRMAN JACKSON: Well, the question becomes, in
19 terms of a metric, on the outcomes, the people who attend
20 the seminars walk away, you feel, with more clarity in that
21 regard? That's a metric.
22 MR. NEWBERRY: More clarity, yes, but my own view
23 is it's going to take continuing attention for a period of
24 time here with on-the-job attention.
25 CHAIRMAN JACKSON: Application and guidance.

1 MR. NEWBERRY: Yes, very definitely. It's only a
2 two-hour seminar.
3 CHAIRMAN JACKSON: Yes, I understand the point.
4 MR. NEWBERRY: Yes.
5 CHAIRMAN JACKSON: But you're at least opening the
6 minds in this regard.
7 MR. NEWBERRY: Yes. They're lively time periods.
8 CHAIRMAN JACKSON: Yes. Okay.
9 MR. BARANOWSKI: In addition to the seminar I just
10 mentioned, the PRA Basics for Regulatory Applications course
11 -- that's course number P-105 -- has been modified to
12 include some additional information on the regulatory
13 approaches for the Reg Guides and the SRP's.
14 That's also a mandatory course for NRR technical
15 staff, and over the next two fiscal years, we would expect
16 that the full staff should be trained, would have attended
17 that particular course. Next viewgraph, please.
18 [Slide.]
19 The resident inspectors' needs are intended to be
20 addressed by the PRA Technology for Regulatory Perspectives
21 course, P-111, which is mandatory for all full-time NRR
22 inspectors and regional reactor program inspectors.
23 Now, the course curriculum includes extensive
24 practical workshops and case studies applicable to the needs
25 of the inspectors as they would perform risk-informed

1 inspections.

2 The first presentation of the course was

3 originally scheduled for October of this year. It was

4 delayed until January of next year as a result of what we

5 call a pilot talk-through, where we sort of try the course

6 out on some more experienced people to see if the concepts

7 that were intended to be in there are coming through

8 clearly.

9 So a few modifications are being made, and then we

10 would expect to have several of these courses in Fiscal Year

11 1998 and finish up in 1999. Resident inspectors will be

12 given the highest priority for attendance in this course.

13 CHAIRMAN JACKSON: Will there be different

14 training for NMSS staff? Has anyone thought about that?

15 MS. FEDERLINE: Yes, we have. You'll see we

16 highlight it in the paper as one of our issues. We're

17 trying to decide right now what systems approach is

18 appropriate for each of our individual regulated systems,

19 and that would somewhat dictate the types of training. We

20 have had training, dedicated courses for performance

21 assessment in the waste disposal area.

22 CHAIRMAN JACKSON: Okay

23 MR. BARANOWSKI: Just to make one final point, we

24 would like to have at least one resident inspector through

25 this training by December of 1998 at each site, and so

1 that's the way that priorities will be worked out. Next
2 viewgraph.
3 [Slide.]
4 At the recommendation of the PRA training focus
5 group, a PRA Technical Managers course was developed and is
6 now being implemented. The course is required for NRR,
7 AEOD, NMSS, and regional technical managers. It's course
8 P-107. The course has been recently updated to include
9 information on the RG's and SRP's, or at least the
10 approaches that we're talking about taking in them, since
11 they're still in draft.
12 We would expect to conduct a number of course in
13 1998, which would get about two-thirds of the agency's
14 technical managers trained and then the balance in Fiscal
15 Year 1999.
16 At the same time, all offices are looking at their
17 technical training needs with regard to risk-informed Reg
18 Guides and SRP's that might result in some additional
19 training or revisions to the courses that I just mentioned.
20 We would expect to enact additional courses in the future as
21 warranted, which I mentioned earlier.
22 I would like to mention that there was a question
23 raised in the SRM regarding a regional representative on the
24 PRA training focus group. There wasn't one on the PRA
25 training focus group. The NRR representative normally

1 provided representation of inspection personnel in terms of
2 PRA training.
3 We have recently added a regional person on there,
4 but I would like to point out that when the PRA training
5 focus group meets, it's not just four people that meet.
6 There are four specific members, but folks from the
7 inspection staff have, in fact, participated in meetings in
8 the past where we've talked about resident and headquarters
9 inspection personnel training in PRA.
10 CHAIRMAN JACKSON: Right. Well, it's very
11 important, which is why the Commission asked for that,
12 because of the fact that you, for instance, talk about
13 having a resident at each site.
14 I mean you need to have a member who has equal
15 weight at the table that can represent the interests,
16 because those are our folks who are out there on a
17 day-to-day basis, interfacing with the licensees, looking at
18 how they handle configuration management, overseeing outage
19 activities, overseeing on-line maintenance, any number of
20 things.
21 Their interests and what they do -- and it is
22 unique to what they do -- need to be represented fully in
23 what you do, not just as visitors to the meetings. So it's
24 a very important issue. Yes.
25 COMMISSIONER DICUS: Do you believe that our

1 training is keeping pace with our other activities in PRA,
2 or is training lagging a little bit behind it?
3 MR. BARANOWSKI: That's a good question.
4 COMMISSIONER DICUS: Or do you feel that it
5 perhaps is, and that's a concern?
6 MR. BARANOWSKI: I guess my personal feeling is
7 that you need to have an understanding of what your
8 regulatory program is before you can train people to execute
9 that program. I wouldn't want to use training as an
10 approach for developing the regulatory program. I've seen
11 some of that tried in some of our meetings, and I don't
12 think it works very well.
13 So it may be we're lagging a little bit, but I
14 think we're addressing things pretty rapidly.
15 CHAIRMAN JACKSON: Once you get the pieces in
16 place.
17 MR. BARANOWSKI: There's a very big push to get
18 this training going.
19 CHAIRMAN JACKSON: Commissioner McGaffigan.
20 COMMISSIONER McGAFFIGAN: How long do these
21 courses last? I mean I'm just trying to get a sense of how
22 much one might possibly take away from it.
23 MR. BARANOWSKI: A typical course is running four
24 or five days. Now, there are some plans to have some
25 two-week courses and things like that. I don't remember the

1 exact length of them.

2 COMMISSIONER McGAFFIGAN: The ones that you have

3 here -- the 107 and 111, et cetera -- they're all four- or

4 five-day?

5 MR. BARANOWSKI: I believe they're four or five

6 days.

7 MR. THADANI: I see three and four days.

8 MR. CALLAN: Three and four days. In classroom

9 hours, something on the order of 18 to 24.

10 MR. BARANOWSKI: We're looking for about 25, I

11 think it was.

12 MR. CALLAN: Twenty-five?

13 COMMISSIONER McGAFFIGAN: I suspect we're going to

14 have to have refresher course.

15 MR. CALLAN: Right. Our concern right now, just

16 to put that in perspective, I think NRR has asked for 400

17 slots of a four-day course, and we're trying to find the

18 money.

19 CHAIRMAN JACKSON: Right. Plus you want to try to

20 get a certain baseline.

21 MR. CALLAN: We're trying to get a baseline, so --

22 I think I know where you're going with the question. It's a

23 survey course -- we understand that -- but it's to

24 accomplish some of the objectives that Commissioner Diaz

25 mentioned and others, to get a baseline. But we are

1 concerned about the resource expenditure it's going to take
2 to do this training, and we haven't solved that problem yet.
3 CHAIRMAN JACKSON: You wanted to make a comment?
4 Could you identify yourself?
5 MR. COE: Yes. I'm Doug Coe with NRI. I just
6 wanted to correct one thing. P-11, the course that's
7 specifically designed for inspectors, will run the better
8 part of two weeks.
9 CHAIRMAN JACKSON: Thank you.
10 MR. BARANOWSKI: And these courses are also in
11 addition to our other more exacting curriculum, which a
12 lesser number of people are taking.
13 CHAIRMAN JACKSON: Right. Let me just ask a
14 question that is somewhat disconnected. How are we creeping
15 up on -- or are we -- performance-based regulation? What
16 are we doing in that regard?
17 MR. THADANI: There are two parts to that effort.
18 Part one is, as we have indicated before, we're trying to
19 utilize some of the thinking that went behind the
20 maintenance rule. Where areas are amenable to risk
21 analysis, we're folding in the performance-based aspects as
22 feedback for any follow-on actions. So that's embedded in
23 what we're doing in terms of areas which are amenable to
24 risk analysis.
25 As far as the other areas, the Commission directed

1 the staff in an SRM to also look at how performance-based
2 approaches could be used in areas which are not amenable to
3 risk analysis.
4 We have a paper due to the Commission, I believe
5 in two weeks, in which we're going to address what is it
6 that we're doing to respond to that Commission direction.
7 Basically, we're going to tie it to integrate with the
8 SI-13, which is the role of industry, and we are planning to
9 have some workshops. But those are actions that are going
10 to come.
11 CHAIRMAN JACKSON: So the more formalized one has
12 to do with this response to the SRM --
13 MR. THADANI: Yes.
14 CHAIRMAN JACKSON: -- and working with industry,
15 and the other is on the more ad hoc basis of utilizing
16 approaches a la the maintenance rule when it seems
17 appropriate.
18 MR. THADANI: That's right. That's right.
19 CHAIRMAN JACKSON: Yes, Commissioner.
20 COMMISSIONER DIAZ: As a parallel, how are we
21 doing with the maintenance rule assessments? I guess we've
22 got now -- what? -- 36?
23 MR. THADANI: Thirty-six, I believe we've
24 completed.
25 COMMISSIONER DIAZ: Thirty-six? Can you tell us,

1 you're happy with the way they're looking at -- the
2 responses?
3 MR. THADANI: Let me ask Scott to give you, up to
4 date, the status.
5 MR. NEWBERRY: I thought we might get that
6 question, so I even conducted a poll of happiness. I think
7 it depends on the expectation. If you were to ask some,
8 they believe there's still a significant way to go. But I
9 think, in the larger picture, we've come a long way in terms
10 of where we were before the maintenance rule, before the use
11 of -- you know, very little use of risk information.
12 Now, you see all plants the use of risk
13 information in terms of qualitative or even very good models
14 and work stations and the like. So I think overall, in the
15 broad context, we've been reasonably pleased with the
16 progress, but, of course, there have been some issues
17 identified.
18 COMMISSIONER DIAZ: And so those 36 plants that we
19 have now -- you know, have assessments completed already --
20 when you say performance-based, they know what it is.
21 MR. NEWBERRY: I can't agree with you. I don't
22 know. I don't know how they would answer that particular
23 question.
24 COMMISSIONER DIAZ: I think Mr. Callan was wanting
25 to speak.

1 MR. CALLAN: The 36 plants, Commissioner, are the
2 plant that have received the baseline programmatic
3 inspection.
4 COMMISSIONER DIAZ: Right.
5 MR. CALLAN: Which, in my view, is not a real good
6 measure or gauge of the maintenance rule, per se. The
7 maintenance rule is not intended to be a programmatic type
8 of rule. And so, as we gain experience with implementing
9 the performance-based aspect of it, then we'll be in a
10 position to, I think, answer your question. We really don't
11 have that much experience to date in the performance-based
12 aspect of the maintenance rule.
13 CHAIRMAN JACKSON: So what you're saying is the
14 baseline inspections have been looking at what licensees
15 have put into place --
16 MR. CALLAN: Right. Yes.
17 CHAIRMAN JACKSON: -- to begin to fully implement
18 the rule. It's not until you have the chance to begin to
19 inspect against their program that you can really address
20 the question.
21 MR. CALLAN: That's right.
22 CHAIRMAN JACKSON: Is that a fair statement?
23 MR. CALLAN: That's right.
24 MR. THADANI: I think we can say one other thing,
25 and that is, when one gets finished with maintenance

1 inspections, you do end up with some understanding of what
2 are those so-called performance measures the industry is
3 going to use, so that that level of confidence is attained.
4 But the actual experience, as Joe correctly noted, we really
5 haven't had.
6 COMMISSIONER DIAZ: But going back to the dead
7 horse or the skeleton, we are needing to define, and the
8 industry understands and we communicate on a one-to-one
9 basis of what it is. I mean it's the expectation that we
10 need to be clear on, even if we're delaying the actual
11 looking at the performance.
12 MR. THADANI: And that was one of the goals behind
13 these programmatic inspections.
14 COMMISSIONER DIAZ: Right.
15 MR. THADANI: It was to make sure that expectation
16 was going to be realized.
17 COMMISSIONER DIAZ: And that's why I asked the
18 question. You know, from these inspections, are we getting
19 a sense that people are moving in the direction of really
20 getting performance measures that can be then "regulated
21 according to the maintenance rule," which is risk-informed
22 performance?
23 MR. THADANI: Yes.
24 COMMISSIONER DIAZ: Okay.
25 MR. BARANOWSKI: Okay. Number 15.

1 [Slide.]

2 Let me briefly mention a few additional things on
3 the risk-based analysis of operating experience. Of course
4 you're aware that we have an agreement now and an SRM of
5 June 13th giving the staff the go-ahead to work on the
6 voluntary approach for obtaining reliability and
7 availability data with the nuclear industry.

8 We recently signed a memorandum of understanding
9 -- or a modification to the memorandum of understanding --
10 which addresses our obtaining that data that we expect to
11 get from industry through INPO, and we should begin
12 receiving that information sometime in Fiscal Year 1998.

13 CHAIRMAN JACKSON: You may recall that, in fact,
14 the staff agreed to characterize for the Commission the
15 scope of the voluntary data arrangement --

16 MR. BARANOWSKI: Yes.

17 CHAIRMAN JACKSON: -- as opposed to the scope of
18 the maintenance rule.

19 MR. BARANOWSKI: Right.

20 CHAIRMAN JACKSON: Now, when will we get that
21 comparative?

22 MR. BARANOWSKI: I would hope you have a
23 memorandum already in your in box.

24 COMMISSIONER McGAFFIGAN: It arrived in our boxes
25 as we were walking downstairs.

1 CHAIRMAN JACKSON: Oh, you did it so you would get
2 us.
3 [Laughter.]
4 MR. BARANOWSKI: It's that kind of trick we pull
5 on you once in a while.
6 COMMISSIONER McGAFFIGAN: It was brought to me as
7 I was walking downstairs.
8 CHAIRMAN JACKSON: Oh, okay.
9 MR. BARANOWSKI: Sorry. We wanted to get it to
10 you a few days early, but we couldn't quite do it.
11 CHAIRMAN JACKSON: Touche.
12 MR. BARANOWSKI: But just in a quick nutshell, the
13 scope of the maintenance rule's systems and components --
14 not structures -- overlaps quite well with the scope of the
15 equipment in the voluntary approach. They're essentially
16 the same. I can't say they're exactly the same.
17 Now, the level of information that's provided on
18 each component or system varies depending on it's perceived
19 risk significance. We tried to put together a more thorough
20 discussion in the paper, and we would be glad to meet either
21 individually or under other circumstances to go over that if
22 necessary.
23 [Laughter.]
24 CHAIRMAN JACKSON: That's very good. Thank you.
25 MR. BARANOWSKI: Let me also mention a few more

1 things. The Accident Sequence Precursor program, we have a
2 paper due to the Commission in November, and so I won't
3 cover too much of that since we're running a little bit
4 late.
5 I will just mention that we've finished the '95
6 work and published that. The '96 precursors have all been
7 identified, and they're going through final QA. Most of
8 them have been finalized and released through the PDR to the
9 public, and we're even into doing some 1997 analyses.
10 In general, what we're finding is about 10 to 15
11 precursors identified per year, and the conditional core
12 damage probabilities run up to a maximum about 10^{-3}, and the
13 10^{-3} we see about once every other year. But we're going to
14 cover this a little bit in our paper, and so I think we'll
15 do a more thorough job in November, if that's okay.
16 CHAIRMAN JACKSON: Sure. And also, there's a
17 question of where are you with respect to developing
18 risk-based performance indicators? Are you going to speak
19 to that in November?
20 MR. BARANOWSKI: I might just mention that the
21 next couple of things that I have listed here, some of the
22 studies that we've done -- reactor core isolation cooling, a
23 special study on fire events, BWR core spray system, and a
24 number of others that we have on progress on auxiliary
25 feedwater systems, reactor protection, initiating events,

1 and loss of off-site power -- are all part of what I would
2 call the ground work for preparing more risk-based
3 performance indicators.
4 I think, after we get a few more of these studies
5 done, we would be wanting to come back to the Commission
6 with some sort of a conceptual idea and see if we're all on
7 the same page on this.
8 CHAIRMAN JACKSON: Okay. Very good.
9 MR. BARANOWSKI: The last thing I'll say is that
10 we're now about to issue our CCF database. It's in a CD-ROM
11 format. It will be issued to the nuclear industry. It does
12 contain proprietary information, and we have worked with
13 INPO to make sure that that can be released to U.S. nuclear
14 power plant operators.
15 Now, I'll turn it back to Tom.
16 MR. KING: All right. Let me try and summarize
17 with the last viewgraph.
18 [Slide.]
19 We've got a lot of things underway or that have
20 been completed over the past six months, but we still have a
21 long way to go. What we've tried to list on 16 were the
22 major things coming up over the next three months or so.
23 As we mentioned, we've got the framework paper
24 from NMSS, which is due in a few weeks. We've got an
25 intense activity to complete the Reg Guides and SRP's.

1 That is going to involve a policy paper to the
2 Commission later this month. It's going to involve some
3 meetings with ACRS -- one next and nne in mid-November -- to
4 go through the comments, the policy issues, and the
5 positions as to how we want to deal with the public comments
6 and finalize the guides.
7 MR. THADANI: Tom, on that, this is a very
8 important point. Tom indicated that a number of significant
9 issues have been raised, and we're going to be meeting with
10 the advisory committee.
11 I think, in view of the significance of some of
12 these issues, we do need to take a week or two to really
13 think through these issues carefully, and so I'm really
14 putting a hedge on whether we can get the paper to the
15 Commission by the end of October. That's a big question
16 mark in my mind.
17 CHAIRMAN JACKSON: Well, you work toward that.
18 MR. THADANI: We are going to work towards this.
19 CHAIRMAN JACKSON: Right? Because, you know --
20 MR. THADANI: Yes.
21 CHAIRMAN JACKSON: -- if you let it slip, things
22 tend to slip forever.
23 MR. THADANI: We're going to work towards it, but
24 I just want to acknowledge that there are some tough issues
25 that we better take a little time to think through.

1 MR. KING: In addition, we've got the pilot
2 activities that we're trying to complete by December, at
3 least in the tech specs and inservice testing area. We've
4 got the ISI package, which is out for comment now, a
5 workshop in November in pilot activities that are just
6 getting underway there.
7 We have the initiative from NEI, where they're
8 going to take a full-scope PRA and compare it against
9 regulatory requirements and operations and maintenance
10 costs, which we hope to finalized and kick off in December.
11 So there's a number of things in the mill. Some of the
12 schedules, as Ashok said, are ambitious, but that's what
13 we're working toward. With that, I conclude.
14 CHAIRMAN JACKSON: Commissioner McGaffigan.
15 COMMISSIONER McGAFFIGAN: I have one question. I
16 apologize. It really goes back to Mr. Newberry's
17 presentation. You said that San Onofrio was out front in
18 terms of the configuration risk management program and that
19 you hoped to have something done by December.
20 Now, do they have a living PRA? And do they
21 calculate, when they take something out, what the
22 conditional core damage frequency is? And do they then
23 adjust for it, take it into account and say, well, that this
24 is too high a risk? And did you even get to the point of
25 discussing what the threshold is for when that might be too

1 high? How did all that work in that discussion?

2 MR. NEWBERRY: In general, they have a very

3 advanced, or a significant program at that plant, such that

4 they would be able to essentially do what you've just

5 suggested there. That's right.

6 And I only wanted to mention that Arkansas was the

7 lead and, in fact, you got the SER, but as that issue

8 unfolded, right now it appears that SONGS is ready to move

9 out on that amendment. That was really the only thing that

10 I wanted to mention.

11 COMMISSIONER McGAFFIGAN: I'm just trying -- is

12 the commitment that they're going to be making something

13 that's going to be captured in a license condition or

14 something?

15 MR. NEWBERRY: A technical specification.

16 COMMISSIONER McGAFFIGAN: In a technical

17 specification.

18 MR. NEWBERRY: In the administrative section, I

19 believe.

20 COMMISSIONER McGAFFIGAN: And can you tell me what

21 number -- is there a number, like, if the conditional core

22 damage frequency approaches some number, then we will think

23 twice about whether we allow the configuration?

24 MR. NEWBERRY: At this point, I don't think there

25 would be a number.

1 COMMISSIONER McGAFFIGAN: So it's a qualitative
2 judgment we're leaving to the licensee.
3 MR. NEWBERRY: Yes, with -- I think there's five
4 or six elements -- just like principles -- that would go
5 into the tech spec considerations.
6 MR. THADANI: I might add to what Scott is saying
7 -- and we're going to be addressing this as one of the
8 issues -- while we have talked about core damage frequency
9 of 10^{-4}, it's an average estimate over a period of one year.
10 COMMISSIONER McGAFFIGAN: Right.
11 MR. THADANI: And the next issue is what kind of
12 instantaneous risk or dynamic aspect of it would one want to
13 consider. That issue we're going to address amongst the --
14 I think that's a policy matter, as well. It needs to be --
15 COMMISSIONER McGAFFIGAN: The reason I'm asking
16 the question -- and I apologize; it's late -- is we're
17 considering the should-to-shall issue in A-3 of the
18 maintenance rule at the moment.
19 CHAIRMAN JACKSON: Right.
20 COMMISSIONER McGAFFIGAN: And to some extent, what
21 you're going through in this negotiation with SONGS or in
22 the other combustion engineering plants is a precursor to
23 what happens when "should" gets changed to "shall" and what
24 do we mean by that?
25 CHAIRMAN JACKSON: Right.

1 COMMISSIONER McGAFFIGAN: So I may just ask the
2 question --
3 CHAIRMAN JACKSON: Well, once "should" is changed
4 to "shall," then they'll be forced to address what, in fact,
5 that means on the ground.
6 COMMISSIONER McGAFFIGAN: Right. But I may just
7 ask separately -- or maybe our TA's may get briefed in more
8 detail about --
9 CHAIRMAN JACKSON: I think that would be good to
10 do a TA briefing on that.
11 MR. NEWBERRY: I'll take the action item to look
12 at that.
13 CHAIRMAN JACKSON: Right.
14 MR. NEWBERRY: And we'll put something together.
15 CHAIRMAN JACKSON: I'm just going to go ahead in
16 the reverse order. Do you have any other comments or
17 questions, Commissioner? And then I'm going to go to
18 Commissioner Dicus.
19 COMMISSIONER DIAZ: Okay. I think I wrote
20 something in here that I think is -- it's going back to
21 philosophy. But having looked at these things for some time
22 and looking at gold-plated or hot-dipped galvanized, we need
23 to remember that, you know, in this case, we've been for
24 some time striving to get to a level of achievement.
25 And in that case I am of the opinion that the

1 better is the enemy of the good and that the best is the
2 enemy of the better, that we need to define what is it that
3 we can do and do it, rather than keep trying to make it
4 gold-plated or otherwise, and I think it is an important
5 step.
6 I would like to ask the staff the next time that
7 we come back -- and these are questions that are not as
8 simple as they sound -- but the first question is, are we
9 convinced -- we, the NRC -- that we are going to be a
10 risk-informed agency? And if that is so, have we permeated
11 the structure so everybody knows that that is a fact?
12 Second is, have we convinced licensees,
13 stakeholder, or anybody that we are going to be a
14 risk-informed agency? Because if we are and we haven't done
15 that, then we have a job to do. Even it it's information,
16 whatever, we need to be doing.
17 And if the answers to these two things are yes --
18 and I do hope they are -- then we go back to the dead horse.
19 It's a matter of defining how good is good and where the
20 process needs to lie. I think we need to move to make
21 something happen rather than keep waiting for further
22 definition. But in that sense, what we make happen has to
23 be enforceable in regulatory space.
24 CHAIRMAN JACKSON: Absolutely.
25 COMMISSIONER DIAZ: Thank you.

1 CHAIRMAN JACKSON: Absolutely. Commissioner
2 Dicus.
3 [No response.]
4 I would like to thank the staff for a very
5 informative and, I'll actually say, enjoyable briefing on
6 the agency's PRA activities. Thank you.
7 We commend you for the progress that you've made
8 to date in what is a sometimes difficult area, but at the
9 same time, we encourage you to continue to improve the
10 process -- and we've heard various comments to that effect
11 -- and to provide appropriate -- for yourselves -- review
12 mechanisms and feedback mechanisms to ensure that the PRA is
13 appropriately understood in a risk-informed framework and
14 appropriately used to make your own efforts
15 performance-based in that sense, to have that feedback so
16 that you focus on outcomes.
17 But I think you've made some long steps forward
18 from where we were two years ago, even though the policy
19 statement was there and there was a PRA Implementation Plan,
20 there's a lot more flesh on the bones, and I think you can
21 be proud of that.
22 Clearly, PRA has become an important tool in
23 support of the regulatory process, a risk-informed process.
24 And so we have to strive to enhance the process where
25 necessary, but always to ensure its consistent use where

1 appropriate.

2 Unless there are any further comments, we're

3 adjourned.

4 [Whereupon, at 11:45 a.m., the briefing was

5 concluded.]

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CERTIFICATE

This is to certify that the attached description of a meeting of the U.S. Nuclear Regulatory Commission entitled:

TITLE OF MEETING: BRIEFING ON PRA IMPLEMENTATION PLAN PUBLIC MEETING

PLACE OF MEETING: Rockville, Maryland

DATE OF MEETING: Wednesday, October 15, 1997

was held as herein appears, is a true and accurate record of the meeting, and that this is the original transcript thereof taken stenographically by me, thereafter reduced to typewriting by me or under the direction of the court reporting company

Transcriber: Jan Del Monte

Reporter: JAN DEL MONTE

United States
Nuclear Regulatory Commission

STATUS UPDATE OF PROBABILISTIC RISK ASSESSMENT IMPLEMENTATION PLAN

Ashok C. Thadani
Office of Executive Director for Operations

Scott F. Newberry
Office of Nuclear Reactor Regulation

Thomas L. King
Office of Nuclear Regulatory Research

Patrick W. Baranowsky
Office for Analysis and Evaluation of Operational Data

October 15, 1997

OVERVIEW

- **Background**
- **Recent Accomplishments**
- **Status updates**
  - **new activities**
  - **status of key activities**
  - **responses to SRMs**
- **Future Activities**

BACKGROUND

- May 1997 - Commission briefed on status of PRA implementation
- May 1997 - SRM regarding use of IPE results in the reactor inspection program issued
- June 1997 - Draft regulatory guides and standard review plans for reactors issued for public comment

BACKGROUND (cont.)

- June 1997 - SRM requesting plans for training NRC staff for implementation of risk-informed regulatory activities

- June 1997 - SRM requesting progress reports regarding efforts to address limitations in reliability and availability data provided to the NRC voluntarily by industry

- July 1997 - PRA Implementation Plan updated

RECENT ACCOMPLISHMENTS

- **Draft risk-informed regulatory guidance (DG-1063 & SRP section 3.9.8) for inservice inspection programs issued for comment**

- **Public workshop held to discuss public comments on draft risk-informed regulatory guides (RG), standard review plans (SRP), and NUREG series report regarding reactor applications**

- **Public comments on NUREG-1560 (IPE insights report) addressed in final version**

- **Training program expanded reflecting increased staff use of PRA**

NEW ACTIVITIES IN PRA IMPLEMENTATION PLAN

- **Review of industry proposal for pilot review of several integrated risk-informed facility changes**

- **Development of standards for PRA**

- **Development of a formal framework for application of risk-informed, performance based or less prescriptive approaches in regulating the use of nuclear material**

- **Evaluation of need for revision to Commission's Safety Goal Policy Statement, including consideration of making core damage frequency a fundamental safety goal**

- **Planning for risk study for low-power and shutdown conditions in reactors**

REG GUIDES AND STANDARD REVIEW PLANS

- **Public workshop held in August**

  - **comments generally supportive**
  - **specific concerns:**
    - **application of quantitative guidelines**
    - **required PRA quality**

- **Formal written comments being reviewed**

- **Policy issues identified:**

  - **treatment of uncertainties**
  - **acceptance guidelines for very small or neglible risk increases**

FOLLOW-UP OF IPE AND IPEEE INSIGHTS

- **IPE Follow-up Actions**

  - analysis of plants appearing to approach quantitative health objectives
  - analysis of plants with relatively high core damage frequencies and conditional containment failure probabilities
  - assessment of selected generic issues
  - follow-up on specific containment performance issues
  - audit of licensee-identified improvements credited in IPE

- **IPEEE**

  - review of interim report on IPEEE results (November 1997)
  - development of follow-up plan similar to that for IPE

PILOT APPLICATIONS

- **Technical Specifications**
  - **Commission approved issuance of ANO risk-informed TS changes**
  - **licensees developing proposed administrative TS for configuration risk management program**
  - **staff review and approval of administrative TS completes pilot**

- **Graded Quality Assurance**
  - **draft safety evaluation prepared for South Texas**
  - **staff recommendation with safety evaluation transmitted to Commission (October 1997)**

PILOT APPLICATIONS (cont.)

- **Inservice Testing of Pumps and Valves**

  - **team evaluated licensees' PRA on-site in July**
  - **several technical issues still under review**
  - **SER being prepared; due December 1997**

- **Inservice Inspection of Piping**

  - **nothing received for review yet; schedule is uncertain**

USE OF PRA IN REACTOR INSPECTION PROGRAM

- **May 28, 1997, SRM requested that staff expedite:**

  - **use of IPE in prioritizing inspection activities**
  - **improvements in Region's ability to use risk insights**
  - **provision of related training**

- **Staff actions to date:**

  - **training lectures in all regions on use of IPE results**
  - **regions staffed with Senior Reactor Analysts**
  - **guidance on use of PRA put in Inspection Manual**
  - **new training course developed especially for inspectors**

- **Upcoming actions:**

  - **fully train inspection staff**
  - **revise core inspection procedures**
  - **produce inspection procedure for Graded QA**

TRAINING FOR RISK-INFORMED REGULATION

- **June 5, 1997 SRM**

  - requested plans for training staff
  - use of new risk-informed RG and SRP
  - overall PRA methods and techniques

- **Training of NRR Technical Staff**

  - seminar on responsibilities of risk-informed regulation
  - NRR technical staff will be required to attend the four day PRA Basics for Regulatory Applications (P-105) course
  - P-105 course has been modified to include the risk-informed RG and SRP
  - will require significant additions to PRA training schedule for both FY 1998 and FY 1999

TRAINING FOR RISK-INFORMED REGULATION (cont.)

- **Training of NRC Reactor Program Inspectors**

  - **inspectors will be required to attend new PRA Technology for Regulatory Perspective (P-111) course**
  - **first presentation of course delayed until January 1997 to accommodate changes identified during course pilot talk-through**
  - **three courses now scheduled for FY 1998 (four in FY 1999)**
  - **goal to have one resident on each site trained by December 1998**

TRAINING FOR RISK-INFORMED REGULATION (cont.)

- **Training for Technical Managers**

  - **PRA for Technical Managers (P-107) course now required for NRR, AEOD, and regional technical managers**
  - **P-107 course has been modified to include risk-informed RG and SRP**
  - **seven courses scheduled in FY 1998. AEOD prepared to train 2/3 of all NRC technical managers in FY 1998**

- **All offices evaluating other PRA training needs for their staff.**

RISK-BASED ANALYSIS OF OPERATING EXPERIENCE

- Agreement reached with industry on reliability and availability data collection. INPO-NRC agreed to revised memorandum of understanding
- 1995 Accident Sequence Precursor report, 1982–1983 Accident Sequence Precursor report, issued
- Reactor Core Isolation Cooling System Reliability Study, Special Study on Fire Events issued
- Draft BWR High Pressure Core Spray System Reliability Study report sent for internal peer review
- Updated CCF database (1980–1995) and associated technical reports prepared for industry-wide distribution

FUTURE ACTIVITIES
(Next three months)

- Transmit framework paper on NMSS uses of risk assessment — 10/97
- Transmit initial IPEEE insights report — 11/97
- Publish final IPE insights report — 11/97
- Conduct workshop on draft ISI RG and SRP — 11/97
- Complete pilot applications of risk-informed regulation
  - technical specifications — 12/97
  - inservice testing - Comanche Peak — 12/97
- Issue final RGs and SRPs — 12/97

POLICY ISSUE
(Information)

October 14, 1997 SECY-97-234

FOR: The Commissioners

FROM: L. Joseph Callan
Executive Director for Operations

SUBJECT: QUARTERLY STATUS FOR THE PROBABILISTIC RISK ASSESSMENT IMPLEMENTATION PLAN

PURPOSE:

This quarterly report presents the status of activities for the Probabilistic Risk Assessment (PRA) Implementation Plan, including the development of risk-informed standards and guidance. The report also serves to provide responses to Staff Requirements Memoranda (Attachment 1) dated May 28, 1997, June 5, 1997, and June 13, 1997, which include, respectively:

(1) actions the staff has taken to expedite (a) the use of IPE results to prioritize inspection activities; (b) improvements in regional capabilities for the use of PRA and risk insights; and (c) provision of related inspector training;

(2) the staff's plans for training NRC staff on (a) the risk-informed regulatory approach(es) contained in the regulatory guidance and standard review plan documents and (b) overall PRA methods and techniques; and

(3) an update on the staff's efforts to work with industry to address shortfalls and limitations in the data on reliability and availability of risk-significant systems to be provided to the staff voluntarily.

CONTACT:
Ashok Thadani, OEDO
415-1705

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 5 WORKING DAYS FROM THE DATE OF THIS PAPER

BACKGROUND:

In a memorandum dated January 3, 1996, from the Executive Director for Operations to Chairman Jackson, the staff committed to submitting quarterly reports on the status of its development of risk-informed standards and guidance. Previous quarterly reports were sent to the Commission on March 26, June 20, and October 11, 1996, and on January 13, April 3, and July 22, 1997. This quarterly report covers the period July 1, 1997 to September 30, 1997.

DISCUSSION:

Attachments 2 and 3 provide this quarter's implementation plan update. Significant achievements in the past quarter include the following:

- The staff incorporated proposed resolutions of the policy, technical, and process issues in drafts of the application-specific Regulatory Guide (RG) and Standard Review Plan (SRP) for inservice inspection (ISI), and discussed these new drafts with the Advisory Committee on Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR). Both the ACRS and the CRGR have reviewed the guidance and concurred in the staff's proposal to issue the guidance for comment by the public. On August 20, 1997, the staff forwarded the draft guidance documents to the Commission and requested its approval for issuing the documents for comment. Commission approval was received in an October 1, 1997, SRM.

  To facilitate solicitation of public comments on the ISI RG and SRP, the staff will conduct a workshop during the comment period to explain the draft documents and answer questions. The workshop will be held late November or early December, at the Marriott Hotel in Bethesda, Maryland.

  In completing the draft RG and SRP for risk-informed inservice inspection, the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the schedule for issuing the final ISI RG and SRP will slip from February 1998 to April 1998.

  For risk-informed ISI programs, the industry had identified three pilot plants that would submit requests for authorization for the use of risk-informed ISI methodology; these applications have not yet been received. In a letter from NEI, dated August 29, 1997, the industry requested to add two additional plants to the list of pilot applications and identified an aggressive schedule for all the pilot plants. The scope for the two new pilot plants is limited to Class-1 piping (primary coolant system piping only). To date, only one application has been submitted by one of the pilot plants with a limited scope RI-ISI program.

  Due to industry's delays in submitting the applications, and the addition of two plants as pilots, the staff is unable to develop an integrated review schedule for the pilot plants at this time. This schedule is contingent on information regarding actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's requests for additional

information (RAIs). The staff continues to hold working meetings with industry to facilitate the development of regulatory guidelines.

- The staff completed ten more maintenance rule baseline inspections, which included inspection of licensee methods for using PRA in maintenance programs and in inspection of safety assessments performed by licensees when removing equipment from service for maintenance in accordance with Paragraph (a)(3) of the Maintenance Rule. As of September 30, 1997, the staff has completed 36 inspections.

- The NRC staff briefed the Commission in May 1997 on the Individual Plant Examination (IPE) insights report, draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance." As a result of this briefing, the staff received a SRM dated May 28, 1997, requesting the staff "to expedite activities in the following areas: (1) using IPE results to prioritize inspection activities; (2) improving regional capabilities for the use of PRA and risk insights; and (3) providing related inspector training." The staff has been active in accomplishing each of the three items as described in item (3) of Attachment 2.

- In a June 5, 1997, Staff Requirements Memorandum (SRM), the Commission requested information on the plans for training NRC staff on 1) the risk-informed regulatory approach(es) contained in the regulatory guidance and Standard Review Plan documents and 2) overall PRA methods and techniques. Attachment 4 provides the staff's response to the SRM. The attachment describes the training that will be necessary to implement the initiatives discussed in the draft RGs and SRPs for risk-informed regulation.

- The staff has developed responses to all the public comments it has received on draft NUREG-1560 and where appropriate, draft NUREG-1560 has been revised. Attached for information are present drafts of the executive summary of NUREG-1560 (Attachment 5) and Appendix C of NUREG-1560 which documents resolution of comments from the public (Attachment 6). The final version of NUREG-1560 will be published in November 1997.

- A draft interim report has been developed that provides preliminary perspectives and summarizes the information presented in the first 24 Individual Plant Examination for External Events (IPEEE) submittals reviewed by the staff. This interim report will be sent to the Commission by the end of November 1997. A summary of the significant preliminary perspectives from the first 24 IPEEE reviews is presented in Attachment 7.

- In an SRM dated June 13, 1997, the Commission requested that the staff periodically report on their efforts to work with industry to address shortfalls and limitations in the data on reliability and availability of risk-significant systems to be provided to the staff voluntarily. The staff's quarterly report on this activity is provided in item (10) of Attachment 2.

- A three-day public workshop was held on August 11-13, 1997, on the following draft Regulatory Guides, Standard Review Plans, and NUREG report:

  - General Guidance (DG-1061 and SRP)
  - Inservice Testing (DG-1062 and SRP)
  - Graded Quality Assurance (DG-1064)
  - Technical Specifications (DG-1065 and SRP), and
  - The Use of PRA in Risk-Informed Applications (NUREG-1602)

  The workshop was well attended by industry representatives. They offered a number of constructive comments, some criticisms, and some suggestions for changing the guidance. Overall, the comments indicated general support for pursuing risk-informed regulation but in a manner which would necessitate modifications to the draft guidance. The significant issues raised at the workshop are summarized in item (1) of Attachment 2.

- In a letter to the NRC dated August 21, 1997, the Nuclear Energy Institute (NEI) made a proposal for three new risk-informed pilot applications of PRA in support of changes to the licensing basis of operating nuclear power plants. The staff met with NEI on September 17, 1997, to discuss the proposal, including potential NRC activities. The pilots would use a full scope PRA to assess risk versus the regulatory requirements and plant operating and maintenance costs. The staff has concluded that, in concept, the initiative is worthwhile and plans to meet with NEI in November to discuss plans for pursuing the initiative.

- In June, 1997, NRC staff met with representatives of the American Society of Mechanical Engineers (ASME) to discuss cooperation with both industry and professional societies to develop new codes and standards, as directed in the SRM on Direction Setting Issue (DSI) 13, dated March 7, 1997 (see SRM in Attachment 1). The development of PRA standards was one subject of this meeting. At the meeting ASME indicated their interest and is convening an ad hoc committee that will have the responsibility to develop such a standard. This committee will be comprised of ASME personnel, NRC staff, national laboratory, academic, and industry personnel.

  A charter for this committee is now being drafted, and will describe the goals and objectives of the committee, committee membership and associated responsibilities, schedules, and milestones. An addition, the charter will include anticipated scope of the standard (e.g., Level 1, 2 and 3 PRA, including core damage accidents initiated by internal and external events during full power operation) and the level of PRA modeling and analysis appropriate for different PRA uses. The Commission will be informed of progress on this development work in the quarterly updates of the PRA Implementation Plan.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objections to its issuance.

for L. Joseph Callan
Executive Director
for Operations

Attachments:
As stated

DISTRIBUTION:
Commissioners
OGC
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OCA
CIO
CFO
EDO
SECY

Attachment 1

Staff Requirements Memoranda
Addressed in October 1997
PRA Implementation Plan Update

Action: (Morrison, RES/
Collins, NRR

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

SECRETARY

Cys: Callan
Jordan
Thompson

May 28, 1997

IN RESPONSE, PLEASE
REFER TO: M970507

Norry
Blaha
Ross, AEOD

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John C. Hoyle, Secretary

SUBJECT: STAFF REQUIREMENTS - BRIEFING ON IPE INSIGHT REPORT, 2:00 P.M., WEDNESDAY, MAY 7, 1997, COMMISSIONERS' CONFERENCE ROOM, ONE WHITE FLINT NORTH, ROCKVILLE, MARYLAND (OPEN TO PUBLIC ATTENDANCE)

The Commission was briefed by the NRC staff on the Individual Plant Examination (IPE) insight report. The Commission asked the staff to expedite activities in the following areas: (1) using IPE results to prioritize inspection activities; (2) improving regional capabilities for the use of PRA and risk insights; and (3) providing related inspector training.

~~(EDO)~~ (RES) (SECY Suspense: ~~TBD~~ 9/30) 9700206

The Commission asked the staff to provide the scope and schedule of activities related to using IPE results to assess regulatory effectiveness in resolving major safety issues. The Commission specifically requested that the staff provide an estimate of the average cost to respond to the Station Blackout rule per person-rem averted in achieving an average reduction in core damage frequency of 2E-5/RY. These activities should be coordinated with the regulatory effectiveness organization.

~~(EDO)~~ (NRR) (SECY Suspense: 6/27/97) 9700207

After the IPE database has been placed on the Internet, the staff should consider allowing licensees to update their IPEs voluntarily to reflect changes in plant configuration.

(RES)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

June 5, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John C. Hoyle, Secretary

SUBJECT: STAFF REQUIREMENTS - SECY-97-077 - DRAFT REGULATORY GUIDES, STANDARD REVIEW PLANS AND NUREG DOCUMENT IN SUPPORT OF RISK INFORMED REGULATION FOR POWER REACTORS

The Commission has approved publication of the draft regulatory guides, standard review plans and NUREG document for a 90-day public comment period.
(EDO) (SECY Suspense: 6/13/97)

The staff should provide the Commission information on its plans for conducting public workshops. The public workshop(s) to be conducted during the public comment period should be of sufficient duration and depth to provide meaningful insights into the approaches described in the documents.

In addition, the staff should provide the Commission information on its plans for training the NRC staff 1) on the risk-informed regulatory approach(es) contained in the regulatory guidance and standard review plan documents and 2) in overall PRA methods and techniques. Particular attention should be given to increasing basic user-level knowledge of PRA methods at the regional level.
(EDO) (SECY Suspense: 9/30/97)

The staff should continue to evaluate the proposed decision criteria and the methods of ensuring conformance to the criteria. The staff should also develop guidance on how to confirm the assumptions and analyses used to justify current licensing basis changes.

SECY NOTE: THIS SRM, SECY-97-077, AND THE COMMISSION VOTING RECORD CONTAINING THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 5 WORKING DAYS FROM THE DATE OF THIS SRM.

In particular, the staff should explore the following areas to add clarity and consistency to the process.

1. The feasibility of assigning assurance levels for conformance to decision criteria.

2. The use of point values for comparisons with decision criteria, without any explicit consideration of uncertainty (i.e., how consideration of uncertainty should be explicitly considered in conjunction with using point values -- for example, use of probability limits).

3. The implications of small increases in core damage frequency (CDF) and large early release frequency (LERF) codified in the guidance documents, as a function of the uncertainty associated with the PRA results.

4. Codifying in the guidance documents the experience gained from the pilots to provide additional guidance on the "increased management attention" process when proposed changes approach the guidelines.

5. Clarifying the distinction between risk-informed and risk-informed, performance-based regulatory approaches.

The staff should continue to pursue the long range goal of improving the overall quality and consistency of PRAs performed by different licensees by promoting high quality standards.

The staff should continue its efforts to complete, in a timely manner, the pilot applications of risk-informed regulation, and to complete the draft regulatory guidance and standard review plan for inservice inspection.

cc: Chairman Jackson
Commissioner Rogers
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
OGC
CIO
CFO
OCA
OIG
Office Directors, Regions, ACRS, ACNW, ASLBP (via E-Mail)
PDR
DCS

Action: Ross, AEOD/
Thadani, RES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICE OF THE
SECRETARY

Cys: Callan
Jordan
Thompson
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Lieberman, OE
Collins, NRR
Halman, ADM
Allison, AEOD
Shelton, IRM
Meyer, ADM

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June 13, 1997

MEMORANDUM TO: L. Joseph Callan
Executive Director for Operations

FROM: John C. Hoyle, Secretary

SUBJECT: STAFF REQUIREMENTS - SECY-97-101 - PROPOSED RULE, 10 CFR SECTION 50.76, "REPORTING RELIABILITY AND AVAILABILITY INFORMATION FOR RISK-SIGNIFICANT SYSTEMS AND EQUIPMENT"

The Commission has approved the staff's recommendation to accept the voluntary approach proposed by NEI for obtaining reliability and availability data for key safety systems.

The staff should continue to work with industry representatives to improve the content of the voluntary data. These improvements should seek to expand the voluntary program to minimize 1) uncertainty in data and 2) the use of compensatory measures to derive needed parameter estimates. The staff should periodically update the Commission on its efforts to work with industry to address shortfalls and limitations in the data, and advise the Commission on whether the voluntary approach remains a viable method of meeting regulatory needs. This periodic update may be incorporated, as appropriate, in the quarterly updates to the PRA implementation plan.

(~~EDO~~) (AEOD/RES) (SECY Suspense: 9/26/97) 9500047, RES

SECY NOTE: THIS SRM, SECY-97-101, AND THE COMMISSION VOTING RECORD CONTAINING THE VOTE SHEETS OF ALL COMMISSIONERS WILL BE MADE PUBLICLY AVAILABLE 5 WORKING DAYS FROM THE DATE OF THIS SRM.

Attachment 2

QUARTERLY STATUS UPDATE OF THE AGENCY-WIDE IMPLEMENTATION PLAN FOR PROBABILISTIC RISK ASSESSMENT (PRA) (from June 30, 1997 to September 30, 1997)

SUMMARY OF SIGNIFICANT PROGRESS

(1) Regulatory Guide (RG) and Standard Review Plan (SRP) Development (Tasks 1.1 and 2.1)

On April 8, 1997, the staff sent to the Commission SECY-97-077, "Draft Regulatory Guides, Standard Review Plans and NUREG Document In Support of Risk-Informed Regulation for Power Reactors." SECY-97-077 requested Commission approval to publish for comment four draft Regulatory Guides (RGs), three draft Standard Review Plan (SRP) sections, and one draft NUREG series report that support implementation of risk-informed regulation for power reactors. By Staff Requirements Memorandum (SRM) dated June 5, 1997, the Commission approved publication of the draft documents. A notice was placed in the Federal Register announcing availability of the documents and requesting public comment on them.

Public Workshop on Regulatory Guides and Standard Review Plans

To facilitate solicitation of public comments, the staff held a workshop on August 11, 12, and 13, 1997, at the DoubleTree Hotel in Rockville, Maryland to explain the draft documents and answer questions. The workshop was well attended by industry representatives. They offered a number of constructive comments, some criticisms, and some suggestions for changing the guidance. Overall, the comments indicated general support for pursuing risk-informed regulation but in a manner which would necessitate some modifications to the draft guidance. The more significant issues raised during the workshop regarding the general regulatory guidance included:

- how the guidelines on CDF and LERF would be applied when proposed increases in risk are very small;
- the conditions under which a full scope PRA would be necessary;
- what constitutes a "quality PRA" and the role of NUREG-1602 in judging the quality of the PRA supporting an application;
- having separate acceptance guidelines for accident sequences initiated during power operation and sequences initiated during low-power and shutdown operations;
- having new industry/NRC pilot programs to ensure the effectiveness of the guidance issued for use.

The staff is reviewing the comments provided at the workshop and those formal written public comments it has received.

Draft Regulatory Guide and Standard Review Plan for Inservice Inspection

The staff completed new drafts of the application-specific RG and SRP for inservice inspection (ISI) and discussed them with senior agency management, the Advisory Committee on Reactor Safeguards (ACRS) and the Committee to Review Generic Requirements (CRGR) in a number of meetings held over the past three months. Both the ACRS and the CRGR have completed their reviews of the guidance and concur with the staff's proposal to issue the guidance for comment by the public. On August 20, 1997, the staff sent to the Commission SECY-97-190, "Draft Regulatory Guide and Standard Review Plan on Risk-Informed Inservice Inspection of Piping." SECY-97-190 requested Commission approval to publish for comment the RG and SRP that supports implementation of risk-informed inservice inspection programs. Commission approval was obtained in an October 1, 1997, SRM. In completing the draft RG and SRP, the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the schedule for issuing the final ISI RG and SRP will slip from February 1998 to April 1998.

(2) Pilot Applications (Task 1.2)

For risk-informed ISI programs, the industry had identified three pilot plants that would submit requests for authorization for the use of risk-informed ISI methodology; these applications have not yet been received. In a letter from NEI, dated August 29, 1997, the industry requested to add two additional plants to the list of pilot applications and identified an aggressive schedule for all the pilot plants. The scope for the two new pilot plants is limited to Class-1 piping (primary coolant system piping only). To date, only one application has been submitted by one of the pilot plants with a limited scope RI-ISI program. Due to industry's delays in submitting the applications, and the addition of two plants as pilots, the staff is unable develop an integrated review schedule for the pilot plants at this time. This schedule is contingent on information regarding the actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's RAIs.

As noted in an August 21, 1997, memorandum to the Commission, completion of the RI-IST pilot plant safety evaluation has been delayed. Nevertheless, between July 14 and 18, 1997, the NRC staff and its contractors reviewed PRA models, backup calculations, and data at Comanche Peak Steam Electric Station (CPSES). The review was conducted as part of the staff's evaluation of Texas Utilities Electric Company's (TUE's) proposed RI-IST program and was aimed at determining whether the CPSES PRA is consistent with the quality and scope guidelines in draft Regulatory Guide DG-1061, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis." While the review team identified some minor weaknesses with the CPSES PRA for the RI-IST application (e.g., missing success paths, limited documentation of human error probabilities, optimistic recovery factors for equipment repair, plant-specific performance data not having already been incorporated into the PRA), the review team feels that these issues can be addressed adequately by the licensee. The staff also identified an area in the calculation of sequence success that needs further clarification. The calculated core damage frequency from the licensee's base PRA will approach 1×10^{-4} per year when external event initiators and shutdown operations are taken into account. Thus, the licensee's proposed

RI-IST program is receiving increased NRC technical and management review in accordance with guidance in DG-1061.

In response to staff questions and concerns, South Texas Project (STP) submitted for staff review another Operational QA Program (OQAP) revision, revised procedures for implementing facets of the graded QA program, a proposed Final Safety Analysis Report (FSAR) revision that would invoke 10CFR50.59 change controls on the GQA implementing procedures, and responses to staff information requests. The staff has prepared a safety evaluation for graded QA based on the reviews performed, which will be sent to the Commission via a separate Commission paper in October 1997.

The staff has been working with the Combustion Engineering Owners Group (CEOG) to develop a TS administrative control for a configuration risk management program (CRMP). The CRMP constitutes the third tier of the three-tiered approach the staff has used in reviewing risk-informed TS allowed outage time (AOT) changes. As discussed in SECY 97-095, the staff is requiring licensees to incorporate a commitment to implement a CRMP in the TS as part of the basis for its approval of risk-informed TS AOT changes. Once the staff reaches agreement with the CEOG on a TS administrative control for the CRMP, and commitments are received from the individual pilot licensees, the staff will issue amendments to the lead plant and the other CE pilot licensees that have review results comparable to those for ANO-2, as discussed in SECY-97-085 and endorsed by the Commission in its May 28, 1997, SRM.

(3) Inspections (Task 1.3)

Significant PRA-related technical support has been provided for the agency's Maintenance Rule baseline inspections. As of September 30, 1997, the staff has performed 36 full inspections. These inspections were performed with the support of experienced staff and contractor personnel trained in the use of PRA, using an inspection procedure that focuses on the inspection and assessment of the relevant PRA-related technical aspects of the NRC-approved industry guideline for implementing the rule (i.e., NUMARC 93-01).

New technical guidance on the use of PRA in the power reactor inspection program has been issued with the revision of Inspection Manual Chapter 2515, Appendix C.

The NRC staff briefed the Commission in May 1997 on the Individual Plant Examination (IPE) insights report, NUREG-1560. As a result of this briefing, the staff received a SRM requesting the staff "to expedite activities in the following areas: (1) using IPE results to prioritize inspection activities; (2) improving regional capabilities for the use of PRA and risk insights; and (3) providing related inspector training." The staff has been active in accomplishing each of the three items as discussed below.

Since June 1995, briefings on IPEs have been made by the Office of Nuclear Regulatory Research (RES) staff to all four regional offices. To date, the majority of IPEs reviewed by the staff have been covered in the briefings. In addition, a detailed briefing (tailored for each region) of the results and insights from NUREG-1560 was presented at each region. These briefings (both types) have been attended by the resident inspectors, regional personnel, and plant inspection teams (where applicable). The briefings have been specifically structured to

aid in prioritizing inspection activities, and to provide guidance on how to use PRA results. In many cases, Senior Reactor Analysts (SRA), on assignment in RES, participated in the preparation and presentation of the briefings as part of their developmental training. Consequently, the SRAs have gained a solid knowledge of the variety of information contained in a PRA.

The briefings on IPEs have provided valuable insights, particularly in plant-specific inspection activities. However, since many of the licensee's IPEs are out of date, inspectors will need to use supplementary information, including current licensee PRAs, as available,[1] to draw appropriate inspection insights. Consequently, plant-specific briefings based on the submitted IPEs have been discontinued. Instead, the SRAs in each Regional office and in NRR, who are now either fully trained and certified, or are in training, will continue to provide ongoing PRA advice for site-specific activities with support from headquarters offices as needed. Regional and headquarters SRA activities include: providing risk-based inspection prioritization, event assessment and inspection follow up, maintenance rule inspection support, inspection procedure guidance development, and maintenance of the SRA homepage on the NRC intranet.

(4) Accident Management (Task 1.9)

The staff review of the IPE submittals included an assessment of licensee responses to the requests in GL 88-20 and NUREG-1335 related to accident management. Based on IPE insights, the staff has not identified any areas where immediate industry actions related to accident management appear necessary. However, the following accident management areas raised in the IPE submittals warrant further staff evaluation:

- Inhibiting ADS in boiling water reactors (BWRs)
- Use of drywell sprays to prevent Mark I containment liner failure
- Preclude terminating injection to the reactor from external sources
- Effectiveness of external reactor vessel cooling

These follow-up items will be addressed in the staff's evaluation of the BWROG Emergency Procedure and Severe Accident Guidelines (EP/SAG) described in SECY-97-132.

(5) Evaluating IPE Insights To Determine Necessary Follow-up Activities (Task 1.10)

As part of finalizing NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," the staff has defined an initial set of follow-up activities. This initial set of activities consists of:

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[1]Since there is no requirement for licensees to submit or update PRAs, actual availability to staff in inspection or other activities is on a case-by-case basis. The staff will investigate options for addressing this issue.

- Additional analysis of plants identified in NUREG-1560 as having risks approaching the Commission's quantitative health objectives (QHO's), based on a preliminary screening analysis. This additional analysis will use updated information and refined methods to make a better comparison with the QHOs;

- Analysis of plants with accident sequence frequencies greater than $1x10^{-5}$ per reactor year and/or conditional containment failure probabilities greater than 0.1. The analysis will evaluate whether these plants have features which merit backfit consideration. This will be done in a manner consistent with the Safety Goal screening assessment in the Commission's Regulatory Analysis Guidelines;

- Analysis of selected generic issues which may merit further staff evaluation, such as:

  - Contributors to station blackout including grid unreliability

  - RCP seal LOCA and its associated contribution to core damage and large early release frequency,

  - Steam generator tube rupture;

- Follow-up on whether the actions licensees stated they were taking as a result of their IPE have, in fact, been taken;

- Follow-up on selected licensee responses to Containment Performance Improvement questions included in GL-88-20, Supplements 1 and 3.

In conjunction with this effort, the staff is developing a plan for audit of licensee-identified improvements credited in IPE analyses, to determine the effectiveness of licensee actions to reduce risk. The schedule for finalizing the list of items and a program plan to address those items is scheduled for completion in November 1997.

In a May 21, 1996, Staff Requirements Memorandum (SRM), the Commission requested that the staff track the regulatory uses of IPE/IPEEE results. Additionally, the Commission noted that consideration should be given to linking the resulting IPE/IPEEE databases together in a single, integrated, coherent program. This task was placed under item 1.10 of the PRA Implementation Plan in the October 1996 update, and a structure and linking process is under development. The staff will discuss the database content in the next quarterly Implementation Plan update. Due to other staff priorities, such as support for pilot applications and risk-informed regulatory documents, and delays in authorizing contract funds, the target schedule for defining uses for risk information, clarifying regulatory use, and assessing methods of data collection has been revised from December 1997 to May 1998.

(6) Methods Development and Demonstration (Task 2.4)

The Seabrook nuclear power plant is participating in the first trial PRA application of ATHEANA (A Technique for Human Event Analysis). ATHEANA is a human reliability analysis (HRA)

method under development in RES which addresses errors of commission as well as omission. It focuses on combinations of performance shaping factors and plant conditions which increase the likelihood of certain human errors. In addition to identifying unsafe acts that will be considered for quantification within the plant PRA model, ATHEANA is showing promise for identifying strategies for improving human reliability.

In response to concerns over the influence of management and organizational factors, safety culture, and downsizing and deregulation on human performance and safe plant operations, RES held a workshop in August 1997 to discuss these issues with nationally and internationally recognized leaders in management and safety issues (including experts from academia, utilities, national laboratories, consulting companies, the NRC, DOE, and NASA). The experts presented papers and results of current research and participated in working sessions on these topics. The products of this workshop will be used to suggest research methods and/or to assess the influences of management and organizational factors, safety culture, and the effects of downsizing and deregulation.

(7) Individual Plant Examination (IPE) and IPE of Externally Initiated Events (IPEEE) Reviews (Task 2.5)

Status of IPE Reviews

The reviews of all 75 original IPE submittals (i.e., not including Browns Ferry, Unit 3) have been completed with a staff evaluation report (SER) issued by RES to NRR for each submittal. With the exception of the Crystal River and Susquehanna IPEs, all IPE submittals have now been found to meet the intent of Generic Letter 88-20. The licensees for Crystal River and Susquehanna plan to submit revised IPEs that would address the staff concerns. It is expected that these two revised IPEs will be submitted to the staff by December 1997 and staff review will be completed by June 1998.

Preliminary review of the recently submitted IPE for Browns Ferry, Unit 3, and responses to a staff request for additional information have been completed. It is expected that RES will issue its SER for the Browns Ferry, Unit 3, IPE by December 1997.

IPE Insights Report

In October 1996, the staff issued draft NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," for public comment. Comments were received from numerous licensees, individuals and other government organizations. Overall, the comments received were positive in nature. The staff has developed responses to all the comments received and, where appropriate, draft NUREG-1560 has been revised. Attached for information are the executive summary of NUREG-1560 (Attachment 5) and Appendix C of NUREG-1560 which documents resolution of comments from the public (Attachment 6). The final version of NUREG-1560 will be published in November 1997.

Status of IPEEE Reviews

Of the 74 expected IPEEE submittals, the staff has received 63; four of which were not complete. Currently, 49 submittals are under various stages of review. Nine additional submittals are expected to be received by the end of December 1997, one by June 1998, and the submittal date of one IPEEE has yet to be determined. The staff will complete all IPEEE reviews and Staff Evaluation Reports (SERs) by June 1999. Similar to the IPE program, the staff will take prompt action should any significant vulnerabilities or safety insights be identified in these reviews.

An interim report has been developed that provides preliminary IPEEE perspectives and summarizes the information presented in the first 24 IPEEE submittals reviewed by the staff. This interim report will be sent to the Commission by the end of November 1997. A summary of the significant preliminary perspectives from the first 24 IPEEE reviews is presented in Attachment 7. In addition, a draft report that summarizes the findings and perspectives from all IPEEE reviews will be sent to the Commission in June 1999 and released for public comment. After receipt and review of comments, the staff will issue the final IPEEE insights report in December 1999.

(8) Risk-Based Trends and Patterns Analysis Task (3.1)

The Common Cause Failure (CCF) database has been updated with events through 1995. The database and its associated technical reports are being provided on CD ROM to all nuclear utilities in accordance with the INPO agreement regarding distribution of NPRDS proprietary data. Initial draft reports for the initiating event update, loss of offsite power study, the auxiliary feedwater system study, and the Westinghouse reactor protection system study have been received by AEOD from its contractor and reviewed with the ACRS. In July 1997, the BWR high pressure core spray system draft report was distributed to NRC staff for internal peer review. Comments have been received and are being incorporated into the final report.

(9) Accident Sequence Precursor (ASP) (Task 3.2)

The last 1996 preliminary ASP analysis has been sent to the licensee for review. Four 1996 final analyses are completed and have been sent to the respective licensees and have been made publicly available. Seven precursor analyses are under licensee or AEOD final review. Events of significance, i.e., those with a conditional core damage probability (CCDP) greater than $1x10^{-4}$, include a Catawba event (loss of offsite power with diesel failure), a Wolf Creek event (frazil icing of the ultimate heat sink), a Seabrook event (long term unavailability of an emergency feedwater turbine-driven pump), a Prairie Island event (loss of offsite power to safeguards buses at both units), and a Haddam Neck event (potential inadequate residual heat removal pump net positive suction head following a medium or large loss of coolant accident).

(10) Compile Operating Experience Data (Task 3.5)

INPO submitted a revision to the Memorandum of Agreement with the NRC regarding access to the EPIX database. The EDO indicated his agreement with minor modifications and sent it to INPO on August 21, 1997, for signature.

The Sequence Coding and Search System (SCSS) conversion from a mainframe computer to a PC-based computer has been completed as well as the beta testing of the new system. Direct access capability to SCSS via the Internet is now functional. Training and direct use for all NRC staff will be implemented by December 31, 1997.

(11) Staff Training (Task 3.6)

A new course, "PRA for Technical Managers," has been added to the curriculum and two presentations were held in FY 1997. This course is designed to provide all levels of staff managers with a basic understanding of PRA methods, strengths, and limitations needed to implement risk-informed, performance-based regulations. Current plans are to present this course seven times in FY 1998 in headquarters.

PRA Level 2 and Level 3 courses have been added to the PRA curriculum. The first presentation of the new PRA Level 2 course, "Accident Progression Analysis," was held in February 1997. This three-day course addresses accident phenomenology under post core damage conditions and development of PRA models for this severe-accident regime. Based on feedback from the first presentation of the course, the course is undergoing significant modification. The PRA Level 3 course, "Accident Consequence Analysis," was "dry-run" in early 1997 and a first presentation was given in September 1997. The three-day course addresses environmental transport of radionuclides and the estimation of offsite consequences from core damage accidents. Current plans are to present each course twice a year.

A new course on external events has been completed. This three-day course addresses external events (such as fires, floods, earthquakes, high winds, and transportation accidents) and the development of external-event PRA models such as those used in the IPEEEs. The first presentation of this course was held August 5-7, 1997.

A new course, "PRA Technology and Regulatory Perspectives", is under development and scheduled for first presentation in January 1998. A pilot presentation of the course was given on September 22-26, 1997. The course was originally scheduled to start in October 1997. However, based on the pilot presentation, further development and refinement of the course necessiated its delay to January 1998. The course will replace the PRA Basics for Regulatory Application course and the Insights Into IPEs course for some basic level users.

In a June 5, 1997, Staff Requirements Memorandum, the Commission requested information on the plans for training NRC staff on 1) the risk-informed regulatory approach(es) contained in the regulatory guidance and Standard Review Plan documents and 2) overall PRA methods and techniques. Attachment 4 of this Commission paper provides the staff's response to the SRM, and describes the training that will be necessary to implement the initiatives discussed in the draft RGs and SRPs for risk-informed regulation.

REVISIONS TO THE EXISTING PRA IMPLEMENTATION PLAN

(1) Risk-Informed Regulatory Guides and Standard Review Plans (Tasks 1.1 & 2.1)

In completing the draft RG and SRP for risk-informed inservice inspection (ISI), the staff has found that a greater than expected effort was required to incorporate all points of view and gain a consensus on draft guidance. With this experience, the staff projects that the final ISI RG and SRP will be delayed from February 1998 to April 1998.

(2) Risk-Informed Pilot Applications (Task 1.2)

Inservice Inspection

With respect to the risk-informed ISI programs, the staff expected but has not received a formal submittal from the three pilot plants (Surry, ANO-2 and Fitzpatrick). Based on an NEI letter, dated August 29, 1997, the staff anticipates receiving pilot applications to implement RI-ISI programs through the winter of 1998. This includes applications from two new pilot plants (ANO-1 and Vermont Yankee). The staff continues to hold public working meetings with the industry and with Virginia Power on the Surry pilot, in anticipation of receipt of a formal application and to facilitate the development of regulatory guidelines.

Due to industry's delays in submitting the application for Fitzpatrick and ANO-2, and the addition of two plants as pilots, the staff is unable to develop an integrated review schedule for the pilot plants at this time. The schedule is contingent on information regarding the actual timing of submittals, the quality of the submittals, and the ability of the pilot plant licensees to commit the resources necessary to respond to the staff's RAIs.

With respect to the EPRI method (EPRI-TR-106706), the staff has not received responses to its RAIs. The EPRI method is used by all of the pilots except Surry. Delays in receiving responses to the staff's RAIs could also impact the schedule for the review of the pilot plants.

Inservice Testing

In a June 17, 1997, memorandum to the Commission, the staff stated that it expected to be able to issue the safety evaluation (SE) on the Comanche Peak RI-IST program in October 1997. The October 1997 completion date for the Comanche Peak SE was based on the assumption that TU Electric Company (TUE) would respond satisfactorily to both the second and final round requests for additional information by August 8, 1997. The staff's final round RAI asked the pilot plant licensee to describe how their proposed RI-IST program comports with the draft RI-IST RG and to explain their rationale for any differences.

In a letter to the NRC dated July 31, 1997 (amended by letter dated September 12, 1997), TUE stated they need additional time to determine how Comanche Peak's RI-IST

program comports with the NRC draft guidance. TUE indicated that the resources required to complete the RAIs are also being used to provide support for (1) the third refueling outage of Comanche Peak Unit 2 in the fall of 1997, and (2) the NRC Maintenance Rule baseline inspection scheduled at Comanche Peak in October 1997. TUE plans to respond to the second and third round RAIs by September 30, 1997. This will delay issuance of the SE on the Comanche Peak RI-IST program until at least late December 1997.

In a letter to the NRC dated August 1, 1997, Arizona Public Service Company (APS) informed the staff that its resources must be diverted from the Palo Verde RI-IST program development effort in order to complete other activities (e.g., the 10 year IST program update and improved technical specification implementation). Therefore, due to the resource constraints and operational priorities discussed above, APS indicated that they will not be in a position to resume supporting the RI-IST implementation effort until mid-1998. At that time, APS will provide the NRC staff with a schedule for responding to the third RAI.

Graded Quality Assurance

Task 1.2 of the PRA Implementation Plan states that the target schedule for completing the graded QA safety evaluation (SE) for STP is July 1997. The South Texas Project (STP) is the only graded QA volunteer plant that submitted a revised graded QA program for staff review and approval. The staff has prepared safety evaluation for the STP program that will be transmitted to the Commission in a separate Commission paper in October 1997. Dialogue with STP on several issues as well as competing priorities for staff resources delayed completion of the safety evaluation from July to September 1997. Staff monitoring of activities at all three volunteer plants (STP, Grand Gulf, and Palo Verde) will continue in order to observe the results of equipment categorization for additional systems, and the results of the application of graded QA controls and to assess the integrity of the corrective action and operational performance feedback programs. This monitoring effort is expected to continue for an extended period (several years) to provide the staff with lessons learned.

For the purposes of the PRA Implementation Plan, this phase of volunteer plant interactions will be considered complete when the GQA RG and inspection procedure (IP) are issued in final form. In the future, the staff will continue to monitor the volunteer plant GQA implementation, gain feedback to revise the RG and IP as warranted, and evaluate GQA implementation strategies for other licensees who choose to pursue GQA. Although issuance of the SER for the STP GQA program is expected in October 1997, the completion date for the GQA pilot application remains March 1998 to reflect the expected schedule for issuance of the final GQA inspection procedure.

New Pilot Applications

In parallel with the NEI initiatives to study the risk and cost of regulated activities (see Task 2.7 "Whole-Plant" Risk Studies), the pilot plants will be submitting license

amendment applications related to diesel generator start time and hydrogen control. Specific schedules will be established when each application is received.

(3) Inspections (Task 1.3)

As discussed just above, work has been initiated on an inspection procedure for Graded Quality Assurance (GQA). However, because of higher priority work on the South Texas GQA program safety evaluation, the schedule for completing a draft inspection procedure for GQA has been changed from September 1997 to December 1997 with completion of the final guide in March 1998.

Due to personnel being reassigned to higher priority activities, such as development of the PRA for Regulatory Applications course, the completion date for the review of core inspection procedures has been revised to October 1997.

(4) Application of IPE/IPEEE to Generic Issue Resolution (Task 1.6)

The completion target for identifying generic issues to be audited and selection of plants to be audited has been revised from "TBD" to December 1997.

(5) "Whole-Plant" Risk Study (Task 2.7-New)

In a letter to the NRC, dated August 21, 1997, the Nuclear Energy Institute (NEI) made a proposal for three new risk-informed pilot applications of PRA in support of changes to the licensing basis of operating nuclear power plants. The staff met with NEI on September 17, 1997, to discuss the proposal, including potential NRC activities. The pilots would use a full scope PRA to assess risk versus the regulatory requirements and plant operating and maintenance costs. The staff has concluded that, in concept, the initiative is worthwhile and plans to meet with NEI in November to discuss plans for pursuing the initiative. In parallel with the "whole-plant " risk studies, the pilot plant licensees will be submitting license amendment applications on issues such as diesel generator start time and hydrogen controls. These items will be tracked under PRA Implementation Plan Task 1.2 "Pilot Applications for Risk-Informed Regulatory Initiatives."

(6) PRA Standards Development (Task 2.8-New)

In June, 1997, NRC staff met with representatives of the American Society of Mechanical Engineers (ASME) to discuss cooperation with both industry and professional societies to develop new codes and standards, as directed in the SRM on Direction Setting Issue (DSI) 13, dated March 7, 1997 (see SRM in Attachment 1). The development of PRA standards was one subject of this meeting. At the meeting ASME indicated their interest and is convening an ad hoc committee that will have the responsibility to develop such a standard. This committee will be comprised of ASME personnel, NRC staff, national laboratory, academic, and industry personnel.

A charter for this committee is now being drafted, and will describe the goals and objectives of the committee, committee membership and associated responsibilities, schedules, and milestones. An addition, the charter will include anticipated scope of the standard (e.g., Level 1, 2 and 3 PRA, including core damage accidents initiated by internal and external events during full power operation) and the level of PRA modeling and analysis appropriate for different PRA uses. The Commission will be informed of progress on this development work in the quarterly updates of the PRA Implementation Plan.

(7) Low Power Shutdown Risk Reevaluation (Task 2.9-New)

RES has been assigned responsibility to further investigate methods for estimating the risk of severe accidents initiated during low power and shutdown operational states. The results of this investigation could include, for example, staff activities such as the development of new analysis methods or performance of experiments.

The staff intends to complete planning for this investigation in FY1988. Consistent with agency resources allocations, defined work will begin in FY1999.

(8) Revision of Safety Goal Policy Statement (Task 2.10-New)

SECY-97-208 discusses a number of issues relating to possible revision of the Safety Goal Policy Statement, including the possible elevation of core damage frequency to a fundamental safety goal. The staff recommended that additional discussions with ACRS be undertaken, with a goal of providing a Commission paper by March 31, 1998, which would include the staff's analysis, conclusions, and recommendations. This item has been inserted into the Implementation Plan; however no specific actions will be taken until the SRM on SECY-97-208 is received.

(9) Risk Based Trends and Patterns (Task 3.1)

The date for the component studies (Task 3.1) has been delayed because the cognizant engineer has been detailed to the Millstone Project. The dates for the systems studies have been delayed due to difficulties in applying models to the various system designs in a manner consistent with the reportability of failures and demands in multiple train systems. The delay in the initiating events update is due to difficulty in interpreting the extent of loss of offsite power and the nature of some initiating events from LERs.

(10) Accident Sequence Precursor Program (Task 3.2)

Schedules for development of low power/shutdown models and external events (earthquake and fire) models for use in the Accident Sequence Precursor Program are currently being revised to reflect NRC staff comments on the initial models and staff assigned to higher priority work.

(11) Risk-Based Performance Indicators (Task 3.4)

The delay in the development and implementation of risk-based performance indicators (Task 3.4) is due to the delays in the component and system studies. The outputs of these tasks serve as basic inputs for risk-based PIs.

(12) Risk Assessment of Material Uses (Task 4.4)

The work for developing PRA methods (Task 4.1) for use in evaluating medical devices containing nuclear material has been subsumed into the larger risk assessment of material uses (Task 4.4). A working group of NRC and Agreement States personnel has been chartered to:

- identify and document a technical basis for a risk-informed approach to the regulation of nuclear byproduct material, and
- develop plans for a graded regulatory approach for nuclear byproduct materials, based on risk information.

There was an initial meeting of the working group in mid-June 1997. Additional meetings were held in July, August, and September 1997 and are expected to continue about monthly through September 1998. Contractor support is planned to be available by November 1997 to assist the working group in its activities. The expected completion date of working group activities is September 1998.

(12) Nuclear Material Licensing and High-Level Waste Issues (Task 4.5)

In the SRM of April 15, 1997, about risk-informed, performance-based regulation (DSI-12) the Commission directed the staff to (1) reexamine its risk-informed, performance-based or risk-informed, less prescriptive (RIPB) approaches with regard to nuclear material licensees and to high-level waste issues, to ensure that the needs of those licensees and those areas receive adequate consideration; (2) review the basis for nuclear materials regulations and processes to identify and prioritize those areas that are or, with minimal additional staff effort and resources, could be made amenable to RIPB regulation; and (3) develop a framework for applying PRA to nuclear material uses, similar to the one developed for reactor regulation (SECY-95-280), where appropriate. In a paper that will be transmitted to the Commission in October 1997, the staff will reexamine preliminarily the RIPB approaches that it has identified in the PRA Implementation Plan, primarily those for nuclear materials licensees and high-level waste issues, but also those for low-level wastes, spent fuel storage facilities, and transportation (the other activities included in the PRA implementation plan). Also, the staff will identify preliminarily other NMSS areas that are or, with minimal resources, can be made amenable to RIPB approaches. Finally, the staff will provide a plan for developing a framework for applying RIPB approaches in NMSS regulation.

REVISED TASK TABLES

Attachment 3 provides updates to reflect the progress and revisions to the PRA Implementation Plan from July 1 to September 30, 1997.

ATTACHMENT -3
REVISED PRA IMPLEMENTATION PLAN
TASK TABLE (September 1997)

1.0 REACTOR REGULATION

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.1 DEVELOP STANDARD REVIEW PLANS FOR RISK-INFORMED REGULATION	Standard review plans for NRC staff to use in risk-informed regulatory decision-making.	* Evaluate available industry guidance. * Develop a broad scope standard review plan (SRP) chapters and a series of application specific standard review plan chapters that correspond to industry initiatives. * These SRPs will be consistent with the Regulatory Guides developed for the industry. * Draft SRPs transmitted to Commission to issue for public comment General IST ISI TS * Issue final SRP General IST ISI TS	 4/97C[2] 4/97C 8/97C 4/97C 12/97 12/97 4/98 12/97	NRR /RES

[1] C = Task Completed

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.2 PILOT APPLICATION FOR RISK-INFORMED REGULATORY INITIATIVES	* Evaluate the PRA methodology and develop staff positions on emerging, risk-informed initiatives, including those associated with: 1. Motor operated valves. 2. IST requirements. 3. ISI requirements. 4. Graded quality assurance. 5. Maintenance Rule. 6. Technical specifications. 6a. Commission Approval 6b. Pilot Amendments Issued 7. Other applications to be identified later. (applications related to diesel generator start times and Hydrogen Control are expected)	* Interface with industry groups. * Evaluation of appropriate documentation (e.g., 10 CFR, SRP, Reg Guides, inspection procedures, and industry codes) to identify elements critical to achieving the intent of existing requirements. * Evaluation of industry proposals. * Evaluation of industry pilot program implementation. * As appropriate, complete pilot reviews and issue staff findings on regulatory requests.	1. 2/96C 2. 12/97 (TUE) TBD (APS) 3. TBD 4. 3/98 5. 9/95C 6a. 5/97C 6b. 12/97	NRR
1.3 INSPECTIONS	* Provide guidance on the use of plant-specific and generic information from IPEs and other plant-specific PRAs.	* Develop IC 9900 technical guidance on the use of PRAs in the power reactor inspection program. * Revise IC 2515 Appendix C on the use of PRAs in the power reactor inspection program. * Propose guidance options for inspection procedures related to 50.59 evaluations and regular maintenance observations. * Review core inspection procedures and propose PRA guidance where needed. * Complete revision to proposed core inspection procedures * Issue draft Graded QA Inspection Procedure for public comment * Issue final Graded QA Inspection Procedure	6/97C 7/97 C 10/97 10/97 12/97 12/97 3/98	NRR
	* Provide PRA training for inspectors. * Provide PRA training for Senior Reactor Analysts (SRA)	* Identify inspector functions which should utilize PRA methods, as input to AEOD/TTD for their development and refinement of PRA training for inspectors. * Develop consolidated/comprehensive 2-3 week PRA for regulatory applications training course. * Conduct training for Maintenance Rule baseline inspections * Conduct training courses according to SRA training programs * Rotational assignments for SRAs to gain working experience	7/96C 10/97 8/96C Ongoing Ongoing	NRR NRR/ AEOD NRR NRR/RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
	* Continue to provide expertise in risk assessment to support regional inspection activities and to communicate inspection program guidance and examples of its implementation.	* Monitor the use of risk in inspection reports. * Develop new methodologies and communicate appropriate uses of risk insights to regional offices. * Update inspection procedures as needed. * Assist regional offices as needed. * Conduct Maintenance Rule baseline inspections	Ongoing 7/98	NRR
1.4 OPERATOR LICENSING	Monitor insights from HRAs and PRAs (including IPEs and IPEEEs) and operating experience to identify possible enhancements for inclusion in planned revisions to guidance for operator licensing activities (initial and requalification)	* Revise the Knowledge and Abilities (K/A) Catalogs (NUREGs 1122 and 1123) to incorporate operating experience and risk insights. * Revise the Examiner Standards (NUREG-1021), as needed, to reflect PRA insights.	8/95C 3/97C	NRR NRR
1.5 EVENT ASSESSMENT	* Continue to conduct quantitative event assessments of reactor events while at-power and during low power and shutdown conditions.	* Continue to evaluate 50.72 events using ASP models.	Ongoing	NRR
	* Assess the desirability and feasibility of conducting quantitative risk assessments on non-power reactor events.	* Define the current use of risk analysis methods and insights in current event assessments. * Assess the feasibility of developing appropriate risk assessment models. * Develop recommendations on the feasibility and desirability of conducting quantitative risk assessments.	TBD	NRR
1.6 EVALUATE USE OF PRA IN RESOLUTION OF GENERIC ISSUES	* Audit the adequacy of licensee analyses in IPEs and IPEEEs to identify plant-specific applicability of generic issues closed out based on IPE and IPEEE programs.	* Identify generic safety issues to be audited. * Select plants to be audited for each issue. * Describe and discuss licensees' analyses supporting issue resolution. * Evaluate results to determine regulatory response; i.e., no action, additional audits, or regulatory action.	12/97 12/97 TBD TBD	NRR/RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.7 REGULATORY EFFECTIVENESS EVALUATION	* Assess the effectiveness of major safety issue resolution efforts for reducing risk to public health and safety.	* Develop process/guidance for assessing regulatory effectiveness. * Apply method to assess reduction in risk. * Evaluate result, effectiveness of rules. * Propose modifications to resolution approaches, as needed (SBO rule implementation and RCP seal issue). * Identify other issues for assessment if appropriate.	ongoing ongoing ongoing ongoing TBD	NRR & RES
1.8 ADVANCED REACTOR REVIEWS	* Continue staff reviews of PRAs for design certification applications.	* Continue to apply current staff review process.	Ongoing	NRR
	* Develop SRP to support review of PRAs for design certification reviews of evolutionary reactors (ABWR and System 80+).	* Develop draft SRP to tech staff for review and concurrence. * Finalize SRP.	6/98 12/99	NRR
	* Develop independent technical analyses and criteria for evaluating industry initiatives and petitions regarding simplification of Emergency Preparedness (EP) regulations.	* Reevaluate risk-based aspects of the technical bases for EP (NUREG-0396) using insights from NUREG-1150, the new source term information from NUREG-1465, and available plant design and PRA information for the passive and evolutionary reactor designs.	12/96C	NRR & RES
1.9 ACCIDENT MANAGEMENT	* Develop generic and plant specific risk insights to support staff audits of utility accidents management (A/M) programs at selected plants.	* Develop plant-specific A/M insights/information for selected plants to serve as a basis for assessing completeness of utility A/M program elements (e.g., severe accident training)	TBD	NRR & RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
1.10 EVALUATING IPE INSIGHTS TO DETERMINE NECESSARY FOLLOW-UP ACTIVITIES	* Use insights from the staff review of IPEs to identify potential safety, policy, and technical issues, to determine an appropriate course of action to resolve these potential issues, and to identify possible safety enhancements.	* Review the report "IPE Program: Perspectives on Reactor Safety and Plant Performance" and identify the initial list of required staff and industry actions (if any), including insights on A/M.	9/97C	NRR & RES
		Finalize list of required staff and industry actions.	11/97	
	* Determine appropriate approach for tracking the regulatory uses of IPE/IPEEE results.	* Audit licensee improvements that were credited in the IPEs to determine effectiveness of licensee actions to reduce risk.	TBD	NRR
		* Define use for information, clarify "regulatory use", and assess the most effective methods for data collection.	5/98	
		* If appropriate, develop approach for linking IPE/IPEEE data bases.	12/98	

2.0 REACTOR SAFETY RESEARCH

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
2.1 DEVELOP REGULATORY GUIDES	Regulatory Guides for industry to use in risk-informed regulation.	* Draft PRA Regulatory Guides transmitted to Commission for approval to issue for public comment. General IST ISI GQA TS * Issue final PRA Regulatory Guides. General IST ISI GQA TS	 C C C C C 12/97 12/97 4/98 12/97 12/97	RES
2.2 TECHNICAL SUPPORT	* Provide technical support to agency users of risk assessment in the form of support for risk-based regulation activities, technical reviews, issue risk assessments, statistical analyses, and develop guidance for agency uses of risk assessment.	* Continue to provide ad hoc technical support to agency PRA users. * Expand the database of PRA models available for staff use, expand the scope of available models to include external event and low power and shutdown accidents, and refine the tools needed to use these models, and continue maintenance and user support for SAPHIRE and MACCS computer codes. * Support agency efforts in reactor safety improvements in former Soviet Union countries.	Continuing Continuing Continuing	RES RES RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
2.3 SUPPORT FOR NRR STANDARD REACTOR PRA REVIEWS	* Modify 10 CFR 52 and develop guidance on the use of updated PRAs beyond design certification (as described in SECY 93-087).	* Develop draft guidance and rule. * Solicit public comment. * Finalize staff guidance and rule.	5/98 11/98 12/99	RES RES RES
2.4 METHODS DEVELOPMENT AND DEMONSTRATION	* Develop, demonstrate, maintain, and ensure the quality of methods for performing, reviewing, and using PRAs and related techniques for existing reactor designs.	* Develop and demonstrate methods for including aging effects in PRAs. * Develop and demonstrate methods for including human errors of commission in PRAs. * Develop and demonstrate methods to incorporate organizational performance into PRAs. * Develop and demonstrate methods for fire risk analysis * Develop and demonstrate methods for assessing reliability/risk of digital systems	9/98 9/98 TBD 9/98 6/99	RES RES RES RES RES
2.5 IPE AND IPEEE REVIEWS	* To evaluate IPE/IEEE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover vulnerabilities; and to document the significant safety insights resulting from IPE/IPEEEs.	* Complete reviews of IPE submittals. * Complete reviews of IPEEE submittals. * Continue regional IPE presentations. * Issue IPE insights report for public comment. * Final IPE insights report * Issue preliminary IPEEE insights report * Issue draft final IPEEE insights report	9/97 6/99 C 10/96C 9/97 11/97 12/99	RES RES RES RES RES RES RES
2.6 GENERIC ISSUES PROGRAM	* To conduct generic safety issue management activities, including prioritization, resolution, and documentation, for issues relating to currently operating reactors, for advanced reactors as appropriate, and for development or revision of associated regulatory and standards instruments.	* Continue to prioritize and resolve generic issues.	Continuing	RES
2.7 NEI INITIATIVE TO CONDUCT "WHOLE PLANT" RISK STUDY	* Review NEI initiative to conduct three pilot "whole plant" risk-informed studies of requirements vs. risk and cost	* Agree on ground rules for study * Complete study	1/98 TBD	RES/NRR TBD
2.8 PRA STANDARDS DEVELOPMENT	* work with industry to develop national consensus standard for PRA scope and quality	* Initiate activity * Finalize standard	9/97C TBD	RES
2.9 LOW POWER AND SHUTDOWN BENCHMARK RISK STUDY	*Collect studies of LP&S risk as a benchmark for assessing the need for further staff activities	*Collect and review existing LP&S risk information (domestic and foreign) *Initiate additional work	9/98 10/98	RES
2.10 SAFETY GOAL REVISION	*Assess need to revise Commission's Safety Goal to make core damage frequency a funamental goal and make other changes	*Initiate discussion with ACRS	TBD	RES

3.0 ANALYSIS AND EVALUATION OF OPERATING EXPERIENCE, AND TRAINING

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office
3.1 RISK-BASED TRENDS AND PATTERNS ANALYSIS	* Use reactor operating experience data to assess the trends and patterns in equipment, systems, initiating events, human performance, and important accident sequence.	* Trend performance of risk-important components. * Trend performance of risk-important systems. * Trend frequency of risk-important initiating events. * Trend human performance for reliability characteristics.	12/98 12/98 3/98 TBD	AEOD
	* Evaluate the effectiveness of licensee actions taken to resolve risk significant safety issues.	* Trend reactor operating experience associated with specific safety issues and assess risk implications as a measure of safety performance.	As Needed	AEOD
	* Develop trending methods and special databases for use in AEOD trending activities and for PRA applications in other NRC offices.	* Develop standard trending and statistical analysis procedures for identified areas for reliability and statistical applications. * Develop special software and databases (e.g. common cause failure) for use in trending analyses and PRA studies.	C CCF-C Periodic updates	AEOD
3.2 ACCIDENT SEQUENCE PRECURSOR (ASP) PROGRAM	* Identify and rank risk significance of operational events.	* Screen and analyze LERs, AITs, IITs, and events identified from other sources to obtain ASP events. * Perform independent review of each ASP analyses. Licensees and NRC staff peer review of each analysis. * Complete quality assurance of Rev. 2 simplified plant specific models. * Complete feasibility study for low power and shutdown models. * Complete initial containment performance and consequence models. * Complete development of the Level 2/3 models * Complete the Rev. 3 simplified plant-specific models. * Complete external event models for fire and earthquake * Complete low power/shutdown models	Ongoing Annual report, Ongoing 3/97C 11/96C C 7/99 11/01 TBD TBD	AEOD AEOD RES RES RES RES RES RES RES
	* Provide supplemental information on plant specific performance.	* Share ASP analyses and insights with other NRC offices and Regions.	Annual rpt	AEOD

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office
3.3 INDUSTRY RISK TRENDS	* Provide a measure of industry risk that is as complete as possible to determine whether risk is increasing, decreasing, or remaining constant over time.	* Develop program plan which integrates NRR, RES, and AEOD activities which use design and operating experience to assess the implied level of risk and how it is changing. * Update plan for risk-based analysis of reactor operating experience * Implement program plan elements which will include plant-specific models and insights from IPEs, component and system reliability data, and other risk-important design and operational data in an integrated frame work to periodically evaluate industry trends.	C 3/98 6/99	AEOD
3.4 RISK-BASED PERFORMANCE INDICATORS	* Establish a comprehensive set of performance indicators and supplementary performance measures which are more closely related to risk and provide both early indication and confirmation of plant performance problems.	* Identify new or improved risk-based PIs which use component and system reliability models & human and organizational performance evaluation methods. * Develop and test candidate PIs/performance measures. * Implement risk-based PIs with Commission approval.	C 9/00 1/01	AEOD
3.5 COMPILE OPERATING EXPERIENCE DATA	* Compile operating experience information in database systems suitable for quantitative reliability and risk analysis applications. Information should be scrutable to the source at the event level to the extent practical and be sufficient for estimating reliability and availability parameters for NRC applications.	* Manage and maintain SCSS and the PI data base, provide oversight and access to NPRDS, obtain INPO's SSPI, compile IPE failure data, collect plant-specific reliability and availability data. * Develop, manage, and maintain agency databases for reliability/availability data (equipment performance, initiating events, CCF, ASP, and human performance data). * Determine need to revise LER rule to eliminate unnecessary and less safety-significant reporting. * Determine need to revise reporting rules and to better capture ASP, CCF, and human performance events. * Publish revised LER rule.	Ongoing Ongoing 6/98 6/98 10/99	AEOD

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
3.6 STAFF TRAINING	* Present PRA curriculum as presently scheduled for FY 1996	* Continue current contracts to present courses as scheduled. * Maintain current reactor technology courses that include PRA insights and applications. * Improve courses via feedback. * Review current PRA course material to ensure consistency with Appendix C.	Ongoing Ongoing Ongoing Complete	AEOD
	* Develop and present Appendix C training courses.	* Prepare course material based on Appendix C. * Present courses on Appendix C.	C C	RES and AEOD
	* Determine staff requirements for training, including analysis of knowledge and skills, needed by the NRC staff.	* Review JTAs performed to date. * Perform representative JTAs for staff positions (JTA Pilot Program). * Evaluate staff training requirements as identified in the PRA Implementation Plan and the Technical Training Needs Survey (Phase 2) and incorporate them into the training requirements analysis. * Analyze the results of the JTA Pilot Program and determine requirements for additional JTAs. * Complete JTAs for other staff positions as needed. * Solicit a review of the proposed training requirements. * Finalize the requirements.	C C C C C C Ongoing	AEOD
	* Revise current PRA curriculum and develop new training program to fulfill identified staff needs.	* Prepare new courses to meet identified needs. * Revise current PRA courses to meet identified needs. * Revise current and New PRA course to include RegGuide and SRP information * Revise current reactor technology courses as necessary to include additional PRA insights and applications.	Ongoing Ongoing 9/97C Ongoing	AEOD
	* Present revised PRA training curriculum.	* Establish contracts for presentation of new PRA curriculum. * Present revised reactor technology courses. * Improve courses based on feedback.	Ongoing Ongoing Ongoing	AEOD

4.0 NUCLEAR MATERIALS AND LOW-LEVEL WASTE SAFETY AND SAFEGUARDS REGULATION

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
4.1 Validate risk analysis methodology developed to assess most likely failure modes and human performance in the use of industrial and medical radiation devices.	* Validate risk analysis methodology developed to assess the relative profile of most likely contributors to misadministration for the gamma stereotactic device (gamma knife).	* Hold a workshop consisting of experts in PRA and HRA to examine existing work and to provide recommendations for further methodological development. * Examine the use of Monte Carlo simulation and its application to relative risk profiling. * Examine the use of expert judgement in developing error rates and consequence measures.	8/94 C 9/95 C 9/95 C	NMSS
	* Continue the development of the relative risk methodology, with the addition of event tree modeling of the brachytherapy remote afterloader.	* Develop functionally based generic event trees.	TBD	RES/ NMSS
	* Extend the application of the methodology and its further development into additional devices, including teletherapy and the pulsed high dose rate afterloader.	*Develop generic risk approaches.	TBD	RES/ NMSS
4.2 Continue use of risk assessment of allowable radiation releases and doses associated with low-level radioactive waste and residual activity.	* Develop decision criteria to support regulatory decision making that incorporates both deterministic and risk-based engineering judgement.	* Conduct enhanced participatory rulemaking to establish radiological criteria for decommissioning nuclear sites; technical support for rulemaking including comprehensive risk based assessment of residual contamination. *Develop guidance for implementing the radiological criteria for license termination.. * Work with DOE and EPA to the extent practicable to develop common approaches, assumptions, and models for evaluating risks and alternative remediation methodologies. (Risk harmonization).	8/94 PR C Final Rule Published 7/97 C 2/98 Ongoing	RES & NMSS
4.3 Develop guidance for the review of risk associated with waste repositories.	* Develop a Branch Technical Position on conducting a Performance Assessment of a LLW disposal facility.	* Solicit public comments * Publish final Branch Technical Position	5/97 C. TBD, Dependent on Resources	NMSS & RES

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
4.4 Risk assessment of material uses.	* Develop and demonstrate a risk assessment for industrial gauges containing cesium-137 and cobalt-60 using PRA and other related techniques. * The assessment should allow for modification based on changes in regulatory requirements. * Use empirical data as much as practicable. * Develop and demonstrate risk assessment methods for application to medical and industrial licensee activities.	* Develop and demonstrate methods for determining the risk associated with industrial gauges containing cesium-137 and cobalt-60. * Final report as NUREG *Working Group with contractor assistance to identify and document a technical basis for a risk-informed approach to the regulation of nuclear byproduct material, and to develop plans for a graded approach to nuclear byproduct material regulation based on risk informaiton.	7/98 10/98	
4.5 Framework for Use of PRA in Regulating Nuclear Materials	* develop a framework for applying PRA to nuclear material uses, similar to the one developed for reactor regulation (SECY-95-280), where appropriate.	*Provide plan for developing Framework *Complete Framework	10/97 TBD	NMSS

5.0 HIGH-LEVEL NUCLEAR WASTE REGULATION

Regulatory Activity	Objectives	Methods	Target Schedule	Lead Office(s)
5.1 REGULATION OF HIGH-LEVEL NUCLEAR WASTE	* Develop guidance for the NRC and CNWRA staffs in the use of PA to evaluate the safety of HLW programs.	* Assist the staff in pre-licensing activities and in license application reviews. * Develop a technical assessment capability in total-system and subsystem PA for use in licensing and pre-licensing reviews. * Combine specialized technical disciplines (earth sciences and engineering) with those of system modelers to improve methodology.	Ongoing	NMSS
	* Identify significant events, processes, and parameters affecting total system performance.	* Perform sensitivity studies of key technical issues using iterative performance assessment (IPA).	Ongoing	NMSS
	* Use PA and PSA methods, results and insights to evaluate proposed changes to regulations governing the potential repository at Yucca Mountain.	* Assist the staff to maintain and to refine the regulatory structure in HLW disposal regulations that pertain to PA. * Apply IPA analyses to advise EPA in its development of a Yucca Mountain regulation * Apply IPA analyses to develop a site-specific regulation for a Yucca Mountain site	Ongoing	NMSS
	* Continue PA activities during interactions with DOE during the pre-licensing phase of repository development, site characterization, and repository design.	* Provide guidance to the DOE on site characterization requirements, ongoing design work, and licensing issues important to the DOE's development of a complete and high-quality license application. * Compare results of NRC's iterative performance assessment to DOE's VA to identify major differences/issues.	Ongoing	NMSS
5.2 APPLY PRA TO SPENT FUEL STORAGE FACILITIES	* Demonstrate methods for PRA of spent fuel storage facilities.	* Prepare user needs letter to RES * Conduct PRA of dry cask storage	4/97C 9/99	RES/NMSS
5.3 CONTINUE USE OF RISK ASSESSMENT IN SUPPORT OF RADIOACTIVE MATERIAL TRANSPORTATION	* Use PRA methods, results, and insights to evaluate regulations governing the transportation of radioactive material.	* Update the database on transportation of radioactive materials for future applications * Revalidate the results of NUREG-0170 for spent fuel shipment risk estimates	End of FY 99 6/99	NMSS

Attachment 4

PRA Training to Support Risk-Informed Regulatory Initiatives

BACKGROUND:

In SECY-97-077, dated April 8, 1997, the staff requested Commission approval to publish risk-informed regulatory guides and standard review plans for public comment. In a June 5, 1997, Staff Requirements Memorandum (SRM), the Commission approved publication of these draft documents and directed the staff to "provide the Commission information on its plans for training NRC staff 1) on the risk-informed regulatory approach(es) contained in the regulatory guidance and standard review plan documents and 2) in overall PRA methods and techniques." The Commission noted that "particular attention should be given to increasing basic user-level knowledge of probabilistic risk assessment (PRA) methods at the regional level." The staff's response to this SRM is provided below.

DISCUSSION:

Risk-informed regulation uses data and insights derived from probabilistic risk analyses to complement and support the traditional engineering analysis approach. To support risk-informed licensing decisions, it is essential that the staff and inspectors be familiar with Commission policy and expectations as well as various aspects of PRA analysis methodologies and results. These aspects include, but are not limited to, strengths and limitations of PRA analysis, the scope of PRA analyses, the use of importance measures, and the effects and sources of uncertainty. Furthermore, the staff also must be familiar with the regulatory framework being established to support risk-informed applications from industry. With these objectives in mind, the staff has designed specific minimum mandatory training programs for technical staff in the Office of Nuclear Reactor Regulation (NRR), inspectors in both NRR and the Regional Offices and NRR, and Regional technical managers that are discussed below.[1]

NRR Technical Staff

All NRR technical staff will be required to attend a newly developed seminar on responsibilities associated with risk-informed regulation. This seminar will orient the staff on the uses risk-informed regulatory initiatives and will be led by a NRR senior manager. The seminar covers the PRA Policy Statement, the scope of risk-informed regulation, staff expectations, responsibilities and acceptance criteria.

[1]With much of the initial focus of risk-informed regulatory activities being on reactor applications, the initial training focus has been on NRR and regional staff. Training programs for managers and technical staff in other offices are still under development.

All NRR technical staff will be required to complete the four day "PRA Basics for Regulatory Applications" (P-105) course or its equivalent.[2] The Technical Training Division (TTD) of the Office of Analysis and Evaluation of Operational Data (AEOD) has updated this course to include information contained in the risk-informed RGs and SRPs. The target schedule for completion of this training is the end of fiscal year 1999. The staff is currently evaluating resource needs to meet this target schedule.

NRR and Region-Based Inspectors

Regional and NRR Inspectors associated with the regulation of power reactors will be required to complete the "PRA Technology and Regulatory Perspectives" (P-111) course. This is a new basic user PRA course targeted to the specific needs of inspectors. The course curriculum includes extensive practical workshops and case studies applicable to the needs of the inspector. The first offering of this course is scheduled to begin in October 1997. Resident inspectors will be given the highest attendance priority with the goal of having at least one resident at every site complete the training by the end of 1998.

NRR and Regional Technical Managers

Regional and NRR Technical Managers associated with the regulation of power reactors will be required to complete the three day "PRA for Technical Managers" course (P-107). TTD has updated this course to include information contained in the risk-informed regulatory guides and standard review plans. Seven sessions of P-107 are scheduled for fiscal year 1998 and will be sufficient to train two thirds of the agency's technical managers. Additionally, sufficient courses will be available during fiscal year 1999 to permit remaining technical managers, associated with the regulation of power reactors, to complete P-107 by the end of fiscal year 1999. Senior management will establish attendance priority as required to support implementation of risk-informed regulatory activities.

Additional Agency-Wide Technical Training

The training plan described above will provide sufficient training to support implementation of the risk-informed RGs and SRPs; but due to resource limitations, it will not provide staff with all of the basic user level courses and prerequisites recommended in NUREG/BR-0228, "Guidance for Professional Development of NRC Staff in Regulatory Risk Analysis." Consequently, if additional PRA training is needed to support specific risk-informed regulatory applications, NRC managers will be expected to define such training for their staff. For advanced users of PRA, the NRC's current PRA training curricula includes eleven advanced technology courses.

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[2]NRR technical staff members who have completed basic user level PRA training within the last three years will be exempted from requirement to complete the P-105 course. To ensure that these staff members receive adequate training on the risk-informed documents, they will be required to receive training based on the newly developed risk-informed modules in addition to the risk-informed regulation seminar.

Attachment 5

Executive Summary
NUREG-1560 (final version)

EXECUTIVE SUMMARY

Introduction

On August 8, 1985, the U.S. Nuclear Regulatory Commission (NRC) issued its "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants" (Federal Register, 50FR32138). That policy statement introduced the Commission's plan to address severe accident issues for existing commercial nuclear power plants.

The Commission formulated an approach for systematic safety examination of plants to study particular accident vulnerabilities and desirable, cost-effective changes to ensure that the plants do not pose any undue risk to public health and safety. To implement this approach, the NRC issued Generic Letter (GL) 88-20 in November 1988, requesting that all licensees perform an Individual Plant Examination (IPE) *"to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission."* The purpose and scope of the IPE effort includes examining internal events occurring at full power, including those initiated by internal flooding. In response, the staff received 75 IPE covering regarding 108 nuclear power plant units. The staff then examined the IPE submittals to determine what the collective IPE results imply about the safety of U.S. nuclear power plants and how the IPE program has affected reactor safety. The following sections summarize the results of the IPE Insights Program examination.

Impact of the IPE Program on Reactor Safety

The primary goal of the IPE Program was for licensees to *"identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements."* However, GL 88-20 did not specifically define what constitutes a vulnerability; hence, the IPEs exhibit considerable diversity in the criteria used to define a vulnerability. The wording used in some submittals is such that it is not always clear whether a licensee is identifying a finding as a "vulnerability" or as some other issue worthy of attention. Therefore, a problem considered to be a vulnerability at one plant may not have been specifically identified as a vulnerability at another plant. In fact, less than half of the licensees actually identified "vulnerabilities" in their IPE submittals; however, nearly all of the licensees identified other areas warranting investigation for potential improvements. Thus, the IPE program has served as a catalyst for further improving the overall safety of nuclear power plants.

Only four licensees with boiling water reactor (BWR) plants and 15 licensees with pressurized water reactor (PWR) plants explicitly stated that their plants had vulnerabilities. The following vulnerabilities were identified by the four BWR licensees with no common vulnerabilities cited:

- failure of water supplies to isolation condensers
- failure to maintain high-pressure coolant injection and reactor core isolation cooling when residual heat removal has failed
- failure to control low-pressure injection during an anticipated transient without scram (ATWS)
- drywell steel shell melt-through as a Mark I containment issue

Some of the PWR licensees identified common vulnerabilities. Some of the vulnerabilities in the submittals include:

- loss of reactor coolant pump (RCP) seals leading to a loss of coolant accident (LOCA)

- design and maintenance problems that reduce turbine-driven auxiliary feedwater pump reliability
- internal flooding caused by component failures
- failure of the operator to switchover from the coolant injection phase to the recirculation phase
- loss of critical switchgear ventilation equipment leading to loss of emergency buses
- need to enhance operator guidance for depressurization during steam generator tube ruptures
- inadequate surveillance of specific valves leading to interfacing system LOCAs (ISLOCA)
- loss of specific electrical buses
- compressed air system failures
- inability to cross-tie buses during loss of power conditions

In addition, almost all of the licensees identified plant improvements to address perceived weaknesses in plant design or operation. (Over 500 proposed improvements were identified by the plants.) Most of these plant improvements are classified as procedural/ operational changes (approximately 45 percent), design/hardware changes (approximately 40 percent), or both. Few of the improvements involve maintenance-related changes. Typically, the procedural or design changes indicate revised training in order to properly implement the actual change. Approximately 45 percent of the plant improvements are identified by the licensees as implemented, with approximately 25 percent ***implemented and credited*** in the IPEs. Other improvements are either planned or under evaluation. Some improvements are associated with other requirements (primarily the station blackout rule) and utility activities. However, although these improvements were not necessarily identified as a result of the IPE, in some cases, the licensee is using the IPE to prioritize the improvements and to support decisions regarding their implementation. The specific improvements vary from plant to plant. However, numerous improvements that had significant impact on plant safety include changes to AC and DC power, coolant injection systems, decay heat removal systems, heating, ventilating and air conditioning, and PWR RCP seals.

IPE Results Perspectives (Core Damage Frequency)

In many ways, the IPE results are consistent with the results of previous NRC and industry risk studies. The IPE results indicate that the plant core damage frequency (CDF) is often determined by many different sequences (in combination), rather than being dominated by a single sequence or failure mechanism. The largest contributors to plant CDF and the dominant failures contributing to those sequences vary considerably among the plants (e.g., some are dominated by LOCAs, while others are dominated by station blackout). However, for most plants, support systems are important to the results because support system failures can result in failures of multiple front-line systems. Further, the support system designs and dependency of front-line systems on support systems vary considerably among the plants. That variation explains much of the variability observed in the IPE results.

Consistent with previous risk studies, the CDFs reported in the IPE submittals are lower, on average, for BWR plants than for PWR plants, as shown in Figure E.1. Although both BWR and PWR results are strongly affected by the support system considerations discussed above, a few key differences between the two types of plants contribute to this tendency for lower BWR CDFs and cause a difference in the relative contributions of the accident sequences to plant CDF. The most significant difference is that BWRs have more injection systems than PWRs and can depressurize more easily to use low-pressure injection (LPI) systems. This gives BWRs a lower average contribution from LOCAs. However, the results for individual plants can vary from this general

trend. As shown in Figure E.1, the CDFs for many BWR plants are actually higher than the CDFs for many PWR plants. The variation in the CDFs is primarily driven by a combination of the following factors:

- plant design differences (primarily in support systems such as cooling water, electrical power, ventilation, and air systems)
- variability in modeling assumptions (including whether the models accounted for alternative accident mitigating systems)
- differences in data values (including human error probabilities) used in quantifying the models.

Table E.1 summarizes the key observations regarding the importance and variability of accident classes commonly modeled and discussed in the IPEs.

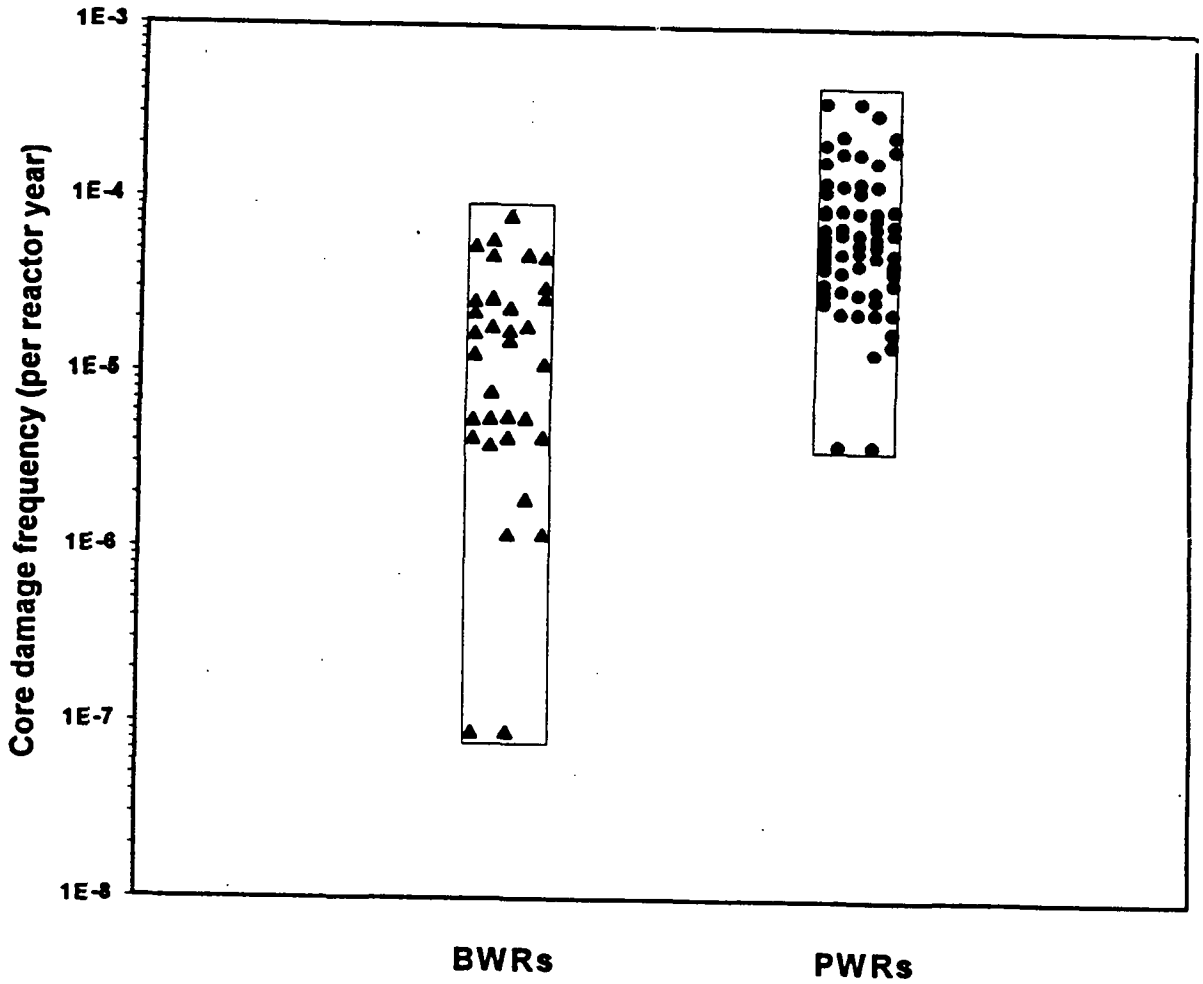

Figure E.1 Summary of BWR and PWR CDFs as reported in the IPEs.

Table E.1 Overview of key CDF observations.

Accident class	Key observations
Transients (other than station blackouts and ATWS)	Important contributor for most plants because of reliance on support systems; failure of such systems can defeat redundancy in front-line systems Both plant-specific design differences and IPE modeling assumptions contribute to variability in results: • use of alternative systems for injection at BWRs • variability in the probability that an operator will fail to depressurize the vessel for LPI in BWRs • availability of an isolation condenser in older BWRs for sequences with loss of decay heat removal (DHR) • susceptibility to harsh environment affecting the availability of coolant injection capability following loss of DHR • capability to use feed-and-bleed cooling for PWRs • susceptibility to RCP seal LOCAs for PWRs • ability to depressurize the reactor coolant system in PWRs affecting the ability to use LPI • ability to cross-tie systems to provide additional redundancy
Station blackouts	Significant contributor for most plants, with variability driven by: • number of redundant and diverse emergency AC power sources • availability of alternative offsite power sources • length of battery life • availability of firewater as a diverse injection system for BWRs • susceptibility to RCP seal LOCAs for PWRs
ATWS	Normally a low contributor to plant CDF because of reliable scram function and successful operator responses BWR variability mostly driven by modeling of human errors and availability of alternative boron injection system PWR variability mostly driven by plant operating characteristics, IPE modeling assumptions, and assessment of the fraction of time the plant has an unfavorable moderator temperature coefficient
Internal floods	Small contributor for most plants because of the separation of systems and compartmentalization in the reactor building, but significant for some because of plant-specific designs Largest contributors involve service water breaks
LOCAs (other than ISLOCAs and steam generator tube ruptures (SGTRs))	Significant contributors for many PWRs with manual switchover to emergency core cooling recirculation mode BWRs generally have lower LOCA CDFs than PWRs for the following reasons: • BWRs have more injection systems • BWRs can more readily depressurize to use low-pressure systems
ISLOCAs	Small contributor to plant CDF for BWRs and PWRs because of the low frequency of initiator Higher relative contribution to early release frequency for PWRs than BWRs because of low early failure frequency from other causes for PWRs
SGTR	Normally a small contributor to CDF for PWRs because of opportunities for the operator to isolate a break and terminate an accident, but important contributor to early release frequency

IPE Results Perspectives (Containment Performance)

For the most part, when the accident progression analyses in the IPEs are viewed globally, they are consistent with typical containment performance analyses. Failure mechanisms identified in the past as being important are also shown to be important in the IPEs. In general, the IPEs confirmed that the large volume PWR containments are more robust than the smaller BWR pressure suppression containments in meeting the challenges of severe accidents; but the IPEs also showed containment bypass was more likely with PWR systems.

Because of the risk importance of early releases, the containment performance analysis descriptions found in the IPE submittals emphasized the phenomena, mechanisms, and accident scenarios that can lead to such releases. These involve early structural failure of the containment, containment bypass, containment isolation failures and, for some BWR plants, deliberate venting of the containment.

As a group, the large dry PWR containments analyzed in the IPEs have significantly smaller conditional probabilities of early structural failure (given core melt) than the BWR pressure suppression containments analyzed. On the other hand, containment bypass and isolation failures are generally more significant for the PWR containments. As seen in Figure E.2, however, these general trends are often not true for individual IPEs because of the considerable range in the results. For instance, conditional probabilities for both early and late containment failure for a number of large dry PWR containments were higher than those reported for some of the BWR pressure suppression containments. Table E.2 summarizes key observations regarding containment performance.

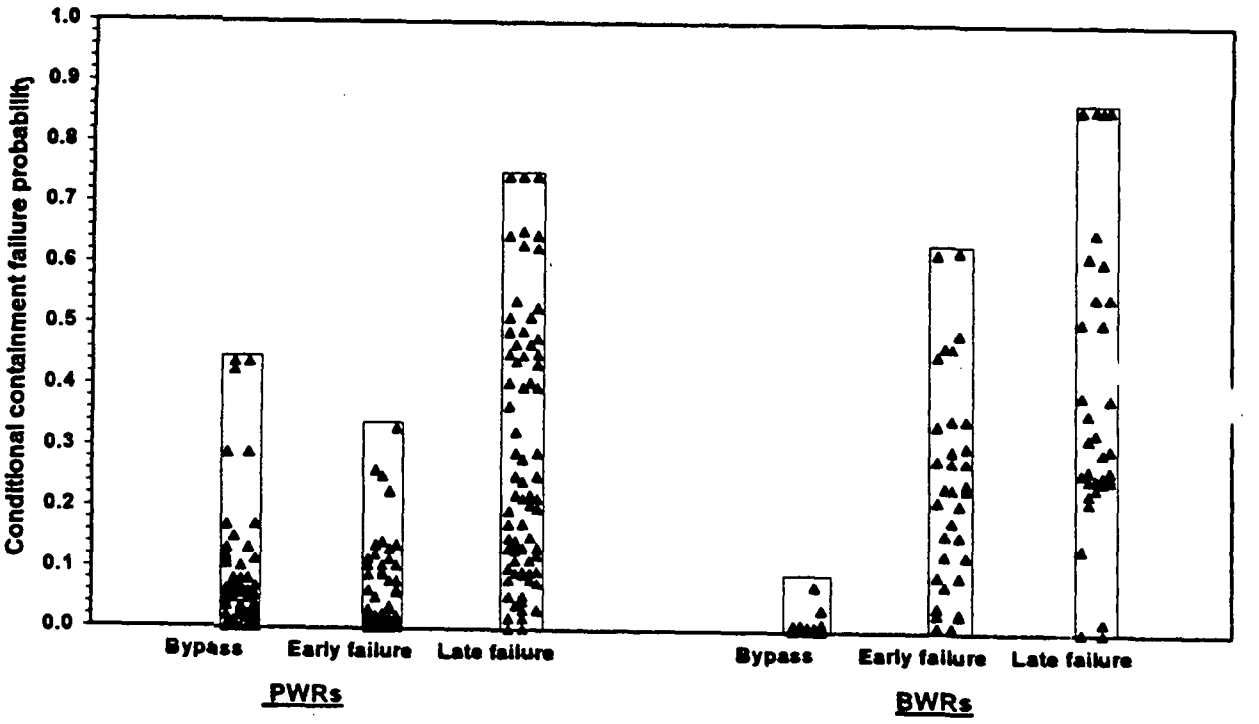

Figure E.2 Summary of conditional containment failure probabilities for BWRs and PWRs as reported in the IPEs.

Table E.2 Key observations regarding containment performance.

Failure mode	Key observations
Early failure	On average, the large volume containments of PWRs are less likely to have early structural failures than the smaller BWR pressure suppression containments Overpressure failures (primarily from ATWS), fuel coolant interaction, and direct impingement of core debris on the containment boundary are important contributors to early failure for BWR containments The higher early structural failures of BWR Mark I containments versus the later BWR containments are driven to a large extent by drywell shell melt-through* In a few BWR analyses, early venting contributes to early releases Phenomena associated with high-pressure melt ejection are the leading causes of early failure for PWR containments* Isolation failures are significant in a number of large, dry and subatmospheric containments The low early failure probabilities for ice condensers relative to the other PWRs appear to be driven by analysis assumptions rather than plant features For both BWR and PWR plants, specific design features lead to a number of unique and significant containment failure modes
Bypass	Probability of bypass is generally higher in PWRs, in part, because of the contribution from SGTRs Bypass, especially SGTR, is an important contributor to early release for PWR containment types Bypass is generally not important for BWRs
Late failure	Overpressurization when containment heat removal is lost is the primary cause of late failure in most PWR and some BWR containments High pressure and temperature loads caused by core-concrete interactions are important for late failure in BWR containments Containment venting is important for avoiding late uncontrolled failure in some Mark I containments The larger volumes of the Mark III containments (relative to Mark I and Mark II containments) are partly responsible for their lower late failure probabilities in comparison to the other BWR containments The likelihood of late failure often depends on the mission times assumed in the analysis

* There has been a significant change in the state-of-knowledge regarding some severe accident phenomena in the time since the IPE analyses were carried out.

The results for BWRs, grouped by containment type, follow expected trends and indicate that, in general, Mark I containments are more likely to fail during a severe accident than the later Mark II and Mark III designs. However, the ranges of predicted failure probabilities are quite large for all BWR containment

designs and there is significant overlap of the results, given core damage. A large variability also exists in the contributions of the different failure modes for each BWR containment group. However, a significant probability of early or late structural failure, given core damage, was found for plants in all three BWR containment groups. The containment performance results for PWRs indicate that most of the containments have relatively low conditional probabilities of early failure, although a large variability exists in the contributions of the different failure modes for both large dry and ice condenser containments.

The results presented i 'he IPE submittals are also consistent with previous studies regarding radionuclide release. A significant early release is of particular concern because of the potential for severe consequences as a result of the short time allowed for radioactivity decay and natural deposition, as well as for accident response actions (such as evacuation of the population in the vicinity of the plant). What is considered to be a significant release varies among the licensees. For many, significant release includes instances involving a release fraction of volatile radionuclides equal to or greater than ten percent of core inventory. Using this definition, the reported conditional probability for significant early release varies from less than 0.01 to 0.5 for the BWR IPEs and from less than 0.01 to 0.3 for the PWR IPEs. Frequencies of significant early release are shown in Figure E.3 and vary from negligible to 2E-5 per reactor year (ry) for BWRs and from 1E-8/ry to 2E-5/ry for PWRs. In the BWR IPEs, significant early releases are almost exclusively caused by early containment failure, while containment bypass (especially SGTR), plays an important role in the reported PWR releases.

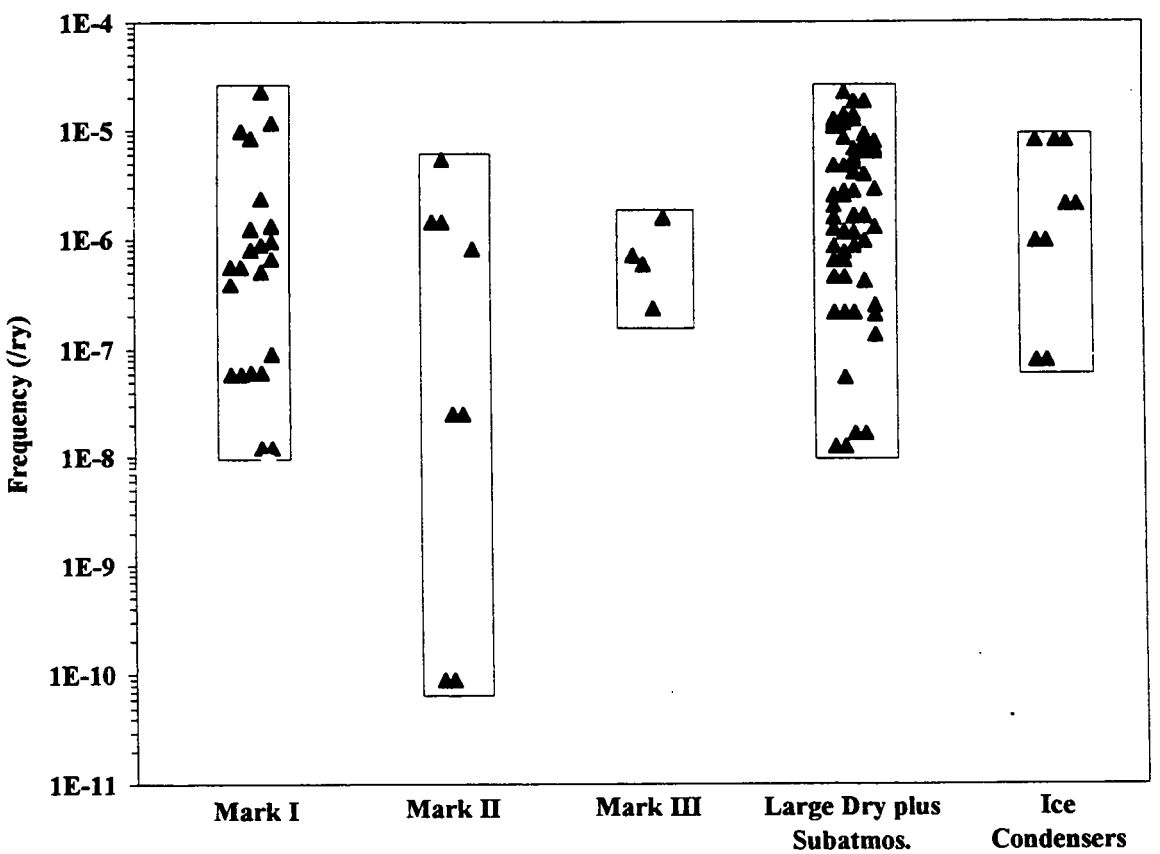

Figure E.3 Frequencies of significant early release (by containment type) as reported in the IPEs.

IPE Results Perspectives (Human Performance)

Only a few specific human actions are consistently important for either BWRs or PWRs as reported in the IPEs. For BWRs, the actions include manual depressurization of the vessel, initiation of standby liquid control during an ATWS, containment venting, and alignment of containment or suppression pool cooling. Manual depressurization of the vessel is more important than expected, because most plant operators are directed by the emergency operating procedures to inhibit the automatic depressurization system (ADS) and, when ADS is inhibited, the operator must manually depressurize the vessel.

Only three human actions are important in more than 50 percent of the IPE submittals for PWRs. These include the switchover to recirculation during LOCAs, initiation of feed-and-bleed, and the actions associated with depressurization and cooldown. Plant-specific features, such as the size of the refueling water storage tank and the degree of automation of the switchover to recirculation, are key in determining the importance of these actions.

While the IPE results indicate that human error can be a significant contributor to CDF, numerous factors can influence the quantification of human error probabilities (HEPs) and introduce significant variability in the resulting HEPs, even for essentially identical actions. General categories of such factors include plant characteristics, modeling details, sequence-specific attributes (e.g., patterns of successes and failures in a given sequence), dependencies, performance shaping factors modeled, application of the human reliability analysis (HRA) method (correctness and thoroughness), and the biases of both the analysts performing the HRA and the plant personnel from whom selected information and judgments are obtained. Although most of these factors introduce appropriate variability in the results (i.e., the derived HEPs reflect "real" differences such as time availability, dependencies and plant-specific factors), several have the potential to cause invalid variability. In order to examine the extent to which variability in the results from the BWRs is caused by valid rather than inappropriate factors, the HEPs from several of the more important human actions appearing in the IPEs were examined across plants.

The results from the examination indicated that much of the observed variability in the HEPs and in the results of the HRAs across IPEs was due to appropriate factors. Thus, the staff concluded that in general the licensees attempted to consider relevant factors in obtaining the HEPs for operator actions and that the results of the HRAs performed by the different licensees were generally consistent, and therefore, useful. On the other hand, it was also concluded that not all of the variability in the examined HEPs was easily explained. That is, after "acceptable" reasons for variation were considered, there still appeared to be some degree of "random" or unexplained variability. Potential reasons for this variability include the lack of precision in existing HRA methods and the fact that some licensees did not perform as thorough HRAs as possible.

While the degree of consistency in HEPs obtained for similar human actions in similar contexts suggests that in general the HRA results from the IPEs were useful in terms of meeting the intent of Generic Letter 88-20, it should be further noted that even when reasonable consistency exists, it is not necessary the case that all the HEPs calculated by a particular plant were realistic and valid for that plant. As noted in Chapters 5 and 13, reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred.

IPE Models and Methods Perspectives

As a result of the IPE program, licensees elected to perform probabilistic risk assessments (PRAs) for their IPEs. Perspectives on the PRAs used in the IPEs provides information on where the models and methods are sophisticated versus where potential

development may be needed. Developing these perspectives requires consideration of several issues:

- What are the characteristics of a PRA?
- How do the IPEs/PRAs compare to these characteristics?
- What can be said about the IPE analyses given the scope of the IPEs (intent of Generic Letter 88-20) and the scope of the staff's IPE reviews?

First, the characteristics of each analytical task of a PRA were defined. The IPE models and methods as described in the individual submittals are compared against the characteristics a PRA. Some general conclusions are then drawn based on the limited reviews that have taken place.

The CDF analyses of the IPEs generally compare well to the characteristics defined for a PRA. The IPEs are generally robust with respect to the identification of dominant accident sequences. This is not to say that particular accident sequence frequencies have been verified, but rather that most of the important accident types have been captured. Although the CDF analyses in the IPEs are generally robust, weaknesses were identified in a number of submittals in the areas of plant-specific data, common cause failure data, and human reliability. An important concern for some IPEs is the HRA, with the most significant concern being the use of invalid HRA assumptions not producing consistently reasonable results.

The IPEs exhibit greater variation in the methods and scope of the accident progression (containment performance) analyses than that found in the CDF analyses. This is commensurate with the guidance of GL 88-20 and NUREG-1335 which allowed significant diversity in the ways licensees could conduct their containment performance analyses. In many of the IPEs, the containment performance analyses and the source term calculations were more simplified than the characteristics identified for a PRA.

Additional IPE Perspectives

The Safety Goal Policy Statement established two qualitative safety goals, which are supported by two quantitative health objectives (QHOs) regarding the risk to the population in the vicinity of a nuclear power plant. Specifically, the safety goals establish that the risk of both prompt fatalities and latent cancer fatalities that might result from reactor accidents should not exceed 0.1 percent of the corresponding risks resulting from all other types of accidents or causes. In responding to GL 88-20, licensees only considered internal events at full power and were not requested to calculate offsite health effects. Therefore, the IPE results cannot be directly compared against the above health objectives. However, it is possible to link the CDF and containment performance results to the safety goals by using surrogate indicators (such as the frequencies of early containment failure and bypass). On the basis of the frequencies of early containment failure and bypass reported in the IPEs, a fraction of the plants have the potential for early fatality risk levels that could approach the QHOs. This subset of plants was further examined using the frequencies of source terms (from the IPEs) with relatively high release factors (>0.03 I, Cs, Te) and adjusted for population. On the basis of this further screening, two BWRs and 12 PWRs remain with the potential for early fatality risk level that could approach the QHOs.

Many of the BWR and PWR plant improvements address station blackout (SBO) concerns and originated as a result of the SBO rule. These improvements had a significant impact in reducing the SBO CDF (an average reduction of approximately 2E-5/ry, as estimated from the CDFs reported by licensees in the IPEs). With the SBO rule implemented, the average SBO CDF is approximately 9E-6/ry, ranging from negligible to approximately 3E-5/ry. Although the majority of the plants that implemented the SBO rule have achieved the goal of limiting the average SBO contribution to core damage to about 1E-5/ry, a few plants are slightly above the goal.

In NUREG-1150, the NRC assessed risk for five nuclear power plants representing both PWR and BWR designs. While these five plants represent only a small sample of designs, it is possible to consider whether the NUREG-1150 results and perspectives are consistent with those found in the IPEs. The average CDFs estimated for both BWRs and PWRs in NUREG-1150 fall within the ranges of the CDFs estimated in the IPEs. The relative contributions of accident sequences in the IPE results is also consistent with the NUREG-1150 results. For PWRs, SBO, transients, and LOCAs are usually the more important contributors; for BWRs, LOCAs and ATWSs are generally less important than SBO and transients.

The conditional probabilities of early containment failure reported in NUREG-1150 (mean values) also fall within the range of the IPE results for each containment type. The IPE results indicate that conditional probabilities for early containment failure are generally higher for BWR containments than for PWR plants. On the basis of absolute frequency, early containment failures for BWRs are similar to those of PWRs because the higher conditional early containment failure probabilities for BWR containments are compensated by the lower values for BWR CDFs.

Overall Conclusions and Observations

In considering the perspectives discussed above, and the results reported in the IPE submittals, certain conclusions and observations can be drawn as summarized below:

- As a result of the IPE program, licensees have generally developed in-house capability with an increased understanding of PRA and severe accidents. Further, the IPE program has served as a catalyst for further improving the overall safety nuclear power plants, and therefore, the generic letter initiative has clearly been a success.

- Areas and issues have been identified where the staff plans to pursue some type of follow-up activity. Areas under consideration are plant improvements, containment performance improvement items either not implemented or not addressed in the IPE submittal, and plants with relatively high CDF or conditional containment failure probability (greater than 1E-4/ry and 0.1, respectively).

- Unresolved safety issue (USI) A-45 ("Shutdown Decay Heat Removal Requirements") and certain other USIs and generic safety issues (GSIs), primarily GSI-23 ("Reactor Coolant Pump Seal Failures"), GSI-105 ("Interfacing System LOCA in Light Water Reactors") and GSI-130 ("Essential Service Water System Failures at Multi-Unit Sites"), were proposed by licensees for resolution on a plant-specific basis. Other safety issues resulting from the IPEs were identified as candidates for further investigation.

- Areas where further research regarding both severe accident behavior and analytical techniques would be useful and should be considered were identified.

- Information from the IPEs/PRAs has the potential to support a diversity of activities such as plant inspection, accident management strategies, maintenance rule implementation, and risk-informed regulation.

- IPE results can be used to indicate areas in PRA where standardization would be useful.

Attachment 6

Appendix C - Public Comments and
NRC Responses on Draft NUREG-1560

APPENDIX C

PUBLIC COMMENTS AND NRC RESPONSES

ON DRAFT NUREG-1560

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ABBREVIATIONS

BWR	Boiling Water Reactor
CCFP	Conditional Containment Failure Probability
CDF	Core Damage Frequency
Cs	Cesium
ECCS	Emergency Core Cooling System
HEP	Human Error Probability
HRA	Human Reliability Analysis
I	Iodine
IPE	Individual Plant Examination
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
MAAP	Modular Accident Analysis Program
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objective
RCP	Reactor Coolant Pump
SER	Staff Evaluation Report
SRV	Safety Relief Valve
SBO	Station Blackout
Te	Tellurium
VP	Vice President

APPENDIX C
PUBLIC COMMENTS AND NRC RESPONSES

C.1 Introduction

NUREG-1560, Volumes 1 and 2 were initially issued in October and November 1996, respectively as a draft report for public comment with the comment period ending May 9, 1997. At that time, notices were published in the *Federal Register* announcing the availability of the report and requesting comment (Ref. C.1). Distribution was made to over 500 people and organizations in the United States and abroad.

To assist readers of the document, a 3-day public workshop was held in April 1997 on the contents of draft NUREG-1560. A notice of this workshop was published in the *Federal Register* (Ref. C.2) and notification of the workshop was sent to all persons receiving the draft report. The workshop took place in Austin, Texas and was attended by representatives of the U. S. Nuclear Regulatory Commission (NRC) and their contractors, representative of the owner's groups, vendors, utilities and their contractors, consultants, and Federal and State agencies. A report summarizing the workshop was prepared and is available for inspection in the NRC Public Document Room (Ref. C.3). The report includes presentation material distributed at the meeting and summarizes the discussion periods during which questions were raised and responses provided. In addition, three sets of written comments were submitted at the meeting. These comments are included in Appendix C or the Workshop Summary Report. The authors and organizations submitting these comments are also listed in Table C.1 (Items #23-26).

In response to the request for comments, the NRC staff received 23 letters. The authors and organizations submitting these letters are listed in Table C.1. All letters received are available for inspection in the NRC Public Document Room.

Table C.1 Submitted comments on draft NUREG-1560.

Identification #	Organization	Author(s)	Date received by NRC
1	Commonwealth Edison Company	Thomas J. Maiman Executive Vice President (VP)	2-14-97
2	Niagara Mohawk	Martin McCormick, Jr. VP Nuclear Engineering	2-14-97
3	South Carolina Electric and Gas Company	Gary J. Taylor VP Nuclear Operations	2-17-97
4	----	Tony Spurgin	2-18-97
5	Centerior Energy	Lew W. Meyers VP	2-25-97
6	Duke Power Company	M.S. Tuckman Sr. VP Nuclear Generation	3-3-97
7	New York Power Authority	James Knubec Chief Nuclear Officer	3-4-97

Table C.1 Submitted comments on draft NUREG-1560.

Identification #	Organization	Author(s)	Date received by NRC
8	Entergy Operations, Inc.	Jerrold G. Dewease VP Operations Support	3-7-97
9	Illinois Power Company	Paul J. Telthorst Director, Licensing	3-7-97
10	Pacific Gas and Electric Company	Gregory Rueger Sr VP & General Manager	3-10-97
11	----	C.A Kukielka, Eric R. Jebsen	3-12-97
12	Carolina Power and Light Company	William Orser Ex.VP Energy Supply	3-14-97
13	PECO Nuclear	G.A. Hunger Director, Licensing	3-14-97
14	TU Electric	C.L. Terry Group VP	3-14-97
15	BWR Owner's Group	----	3-14-97
16	Westinghouse Owner's Group	Louis F. Liberatori, Jr. Vice-Chairman	3-25-97
17	Northeast Utilities Services Company	Sunil Weerokkody Supervisor, PRA	4-10-97
18	GPU Nuclear, Inc.	J.C. Fornicola Director, Licensing & Regulatory Affairs	4-29-97
19	Baltimore Gas and Electric Company	Charles H. Cruse VP Nuclear Energy	3-27-97 5-8-97
20	Public Service Electric and Gas Company	D.R. Powell Manager, Licensing & Regulation	5-9-97
21	IES Utilities, Inc.	John F. Franz VP Nuclear	5-9-97
22	Nuclear Energy Institute	Anthony Pietrangelo Director, Licensing Nuclear Generation	5-9-97
23	Environmental Protection Agency	T. Margulies	*
24	Virginia Power	K. Tuley	*
25	New York State Department of Health	J. Dunkleberger	*
26	NRC-IPE Workshop	**	*

* Written comments submitted at NRC-IPE workshop.
**Verbal comments discussed at NRC-IPE workshop.

In addition to these reviews and comments, as part of the normal review process, the staff discussed the approach and results of draft NUREG-1560 with the Advisory Committee on Reactor Safeguards on several occasions (Ref. C.4).

As discussed in Chapter 1 of this NUREG, the report is comprised of two volumes, with Volume 1 as a summary of the more detailed information contained in Volume 2. However, due to the nature of the comments received on the draft, some of the chapters were rearranged or renamed in the final report. Table C.2 shows the relationship of the draft report to the final report on a chapter by chapter basis. The comments received were reviewed and categorized according to the various chapters. Comments related to the "summary" chapter (from Volume 1) and the associated detailed chapter(s) (from Volume 2) are grouped together.

Table C.2 Relationship of draft NUREG-1560 to the final NUREG-1560

Volume 1 chapters			Volume 2 corresponding detailed chapters	
Final report	**Draft report**		**Final report**	**draft report**
1. Introduction	same	⇒	no corresponding chapter	no corresponding chapter
2. Impact of the IPE Program on Reactor Safety	same	⇒	9. Plant Vulnerabilities and Plant Improvements	same
no corresponding chapter	no corresponding chapter	⇒	10. Background for Obtaining IPE Results Perspectives	10. Background for Obtaining Reactor and Containment Design Perspectives
3. IPE Results Perspectives: Core Damage Frequency	3. Core Damage Frequency Perspectives	⇒	11. IPE Core Damage Frequency Perspectives	11. Reactor Design Perspectives
4. IPE Results Perspectives: Containment Performance	4. Containment Performance Perspectives	⇒	12. IPE Containment Performance Perspectives	12. Containment Design Perspectives
5. IPE Results Perspectives: Human Performance	5. Human Action Perspectives	⇒	13. IPE Human Performance Perspectives	13. Operational Perspectives
6. IPE Models and Methods Perspectives	6. IPEs with Respect to Risk-Informed Regulation	⇒	14. Perspectives on PRA Models and Methods Used in the IPEs	14. Attributes of a Quality PRA
				15. Comparison of IPEs to a Quality PRA
7. Additional IPE Perspectives	same	⇒	15. Safety Goal Implications	16. Safety Goal Implications
			16. Impact of Station Blackout Rule on Core Damage Frequencies	17. Impact of Station Blackout Rule on Core Damage Frequencies
			17. Comparison with NUREG-1150 Perspectives	18. Comparison with NUREG-1150 Perspectives
8. Overall Conclusions and Observations	same	⇒	no corresponding chapter	no corresponding chapter

All of the written comments sent directly to the NRC (Items 1-22 in Table C.1) and submitted at the workshop (Items 23-25 in Table C.1) together with all of the verbal comments provided at the workshop (Item 26 in Table C.1) have been addressed in the final version of NUREG-1560. The comments fell into three broad categories:

(1) A number of comments either were editorial in nature or address the accuracy of the information provided in draft NUREG-1560. For these comments, corrections were made to the text where appropriate. These comments are not reproduced in this appendix with staff response. The comments are available in the NRC Public Document Room.

(2) Some comments were observations in nature and did not appear to solicit a response nor seek a revision to the text of the report. These comments are also not reproduced in this appendix with staff response. The comments are available in the NRC Public Document Room.

(3) Other comments address insights, interpretations and perspectives drawn in the draft NUREG-1560. In some cases, the commentors were concerned that the conclusions were unsubstantiated. In other cases, commentors were concerned about policy implications. For these comments, summaries were developed that captured the concern and an NRC staff response to the comment is provided. These comments and associated responses are provided in the following sections. The specific comments are available in the NRC Public Document Room.

Some of the comments discussed in the following sections are more general in nature and applied to insights, interpretations, etc. discussed in more than one chapter of the report. Comments of this nature can, therefore, appear in several sections of this appendix. An attempt is made in each section to identify those comments that apply to other parts of the NUREG.

C.2 Chapters 2 and 9: Impact of the IPE Program on Reactor Safety

In addition to comments identifying factual errors in these chapters which were corrected, the following general comments were received. These comments and the NRC response are provided below.

1. **Comment:** Numerous erroneous claims of general applicability of vulnerabilities are made in the report. Implying generic applicability of vulnerabilities is inconsistent with the Individual Plant Examination (IPE) purpose which is to identify plant-specific vulnerabilities and cost-effective improvements. (Reference: see Table C.1, #8, 15, 20, 22)

   Response:

   It is true that the generic applicability of identified vulnerabilities cannot be ascertained. In addition, there is no consistent definition of vulnerability used in the IPEs. Further, variability in plant design and operation, as well as different modeling assumptions, can make a vulnerability unique to a particular plant. Therefore, statements regarding generic applicability of vulnerabilities have been rephrased in the NUREG. The purpose of presenting the vulnerabilities and associated plant improvements identified by the licensees is so that all of the licensees may benefit from considering these enhancements as means of improving the safety at their plant in a cost-effective manner.

2. **Comment:** Claims that plant improvements identified by one licensee could be implemented by other plants should not be made. Plant improvements should not be implemented without a full assessment of induced competing risks and the expenditure of resources required that may far outweigh any safety benefit gained. (Reference: see Table C.1, #15)

Response:

All statements about generic application of plant improvements have been rephrased in the NUREG. As with the identification of vulnerabilities, the purpose for discussing identified plant improvements is so that all licensees can benefit by considering their potential implementation at their plant to improve plant safety. A prudent evaluation by a licensee of the benefit of plant improvements identified by other plants would involve both cost-benefit and competing risk considerations.

3. **Comment:** Listing improvement implementation by the licensees as of the date of the IPE submittal is misleading because many plant changes have occurred since the initial IPE submittals. (Reference: see Table C.1, #1, 16)

Response:

NUREG-1560 represents a snapshot in time as far as risk and identified vulnerabilities and plant improvements (including their implementation). It is recognized that many licensees have updated their IPEs and the current status of identified plant improvements may be different than from what was reported in the original submittal. Updated plant improvement status reports are presented in Appendix B for those licensees who provided updated status information in response to the solicitation of comments on Draft NUREG-1560.

C.3 Chapters 3 and 11: IPE Results Perspectives: Core Damage Frequency

Many comments were received concerning the accuracy of the information provided in these chapters or the insights that were identified. Corrections were made to the text where appropriate. In addition, several general comments were provided on the content of this chapter. These comments and an associated response are provided below.

1. **Comment:** The reported core damage frequencies (CDFs) and dominant contributors do not reflect updated probabilistic risk assessment (PRA) results. Many utilities have updated their PRAs one or more times in response to plant design and procedure changes. In addition, many licensees have provided the NRC with revised IPE submittals some with extensive modeling changes and changes in the risk contributors and CDF. To correctly reflect insights from the IPEs requires consideration of supplementary submittals as well. (Reference: see Table C.1, #1, 12, 15, 22)

Response:

Because many plant PRAs are being constantly updated to reflect the current plant design and operation, it is not practical to constantly update NUREG-1560 to incorporate new insights. NUREG-1560 is, and will remain, a compilation of the calculated CDFs and insights obtained from the original IPE submittals. However, information from updated IPE submittals is provided in Appendix B.

2 **Comment:** In comparing the plants, the categorization of boiling water reactors (BWRs) solely by vintage, pressurized water reactors (PWRs) by nuclear steam supply system (NSSS) vendor, and Westinghouse PWRs by the number of loops is not appropriate and can lead to misinterpretation of results. It would be valuable to also look at the results based on a categorization of architect/engineer and/or builder and also age of plant to see if variations can be explained within each NSSS category. Further subgrouping of plants according to similar design characteristics (e.g., emergency core cooling system, ECCS, designs) could be possible. (Reference: see Table C.1, #16)

Response:

Early in the IPE Insights Program, the plants were grouped by architect/engineer and the IPE CDFs within and among these groups were

compared. It was found that comparison of results on this basis was not productive because there is considerable design variability even among plants designed by the same architect/engineer. A decision was made to perform the analysis using plant groups based upon the NSSS vendor to account for basic NSSS design differences. The BWRs were further sub-categorized by vintage to account for differences in ECCS design. The Westinghouse plants were grouped according to the number of loops since the ECCS and other general plant features for the plants in each of these groups are generally the same (see Table 10.3). It is recognized that the balance of plant including support systems for plants in each of the designated groups can be different and skew any comparison of the results for a plant group. The NUREG identifies that these plant-specific features impact the results and draws the appropriate conclusions on the resulting insights. Finally, it is recognized that further subcategorization of plants according to a selected parameter could be made. However, variability in other parameters would likely impact that comparison. Because of this fact and also due to resource limitations, further subcategorization was not pursued.

3. **Comment:** The degree to which a search for variability associated with plant design differences has been made is questionable. The NUREG states that important design features, operator actions, and model assumptions all impact the variability in results. However, few model assumptions are identified. As is well known, substantial differences in PRA results occur because of balance-of-plant and support system design differences despite similarities in NSSS design. Therefore, it is judged that there is no basis to assert that the basis for observed variability is anything but dominated by plant differences in design, procedures, and training. (Reference: see Table C.1, #15)

Response:

Whether plant-specific design/operational differences or modeling assumptions are dominant factors in explaining the variability is not always obvious. However, it is believed that either or both can play a significant role in the variability for certain accident types. In many cases, a judgment is made in the NUREG on which is the dominant factor for an accident class for a plant group. The NUREG identifies that a significant amount of variability is due to support system and other plant-specific design/operational differences. Many of these design/operational differences are highlighted in the report. However, it is also clear that modeling assumptions play an important part in the variability. In some cases, because of limited documentation in the IPE submittals, it is not clear if the modeling assumption really reflects a design or operational difference. For example, many licensees did not credit an alternate coolant injection system because they did not perform an analysis of whether or not it would be successful. The neglect of the potential use of this system is a model assumption until it is shown that, because of plant-specific factors, such a system could not be used. For other cases, it is clear that a model assumption is being made. For example, many licensees assumed that the DC bus load shedding would always successfully occur during a station blackout.

4. **Comment:** The choice of success criteria has a major impact on the variability of the CDF results in a given category of plants. This is not mentioned in the NUREG. Some utilities working with smaller PRA vendors had more stringent success (i.e., conservative) criteria than others who worked with reactor vendors and had access to information that allowed for less conservative success criteria. Also, some larger utilities had the resources to perform the necessary analyses to establish a less conservative success criteria where other utilities did not have such resources and chose to use a conservative success criteria. (Reference: see Table C.1, #16)

Response:

The NUREG identifies where success criteria assumptions impact the variability of the calculated CDFs. As mentioned in the response to the previous comment, because of limited documentation in the submittals, it was not always clear if differences in success criteria were due to design differences or modeling assumptions. The basis for not crediting a system (and in some cases, for crediting a system) or for the operating requirements of a credited system (including support system requirements) were not always documented in the submittals. The CDF evaluation thus made no attempt to validate the differences in success criteria but simply reported its impact on the variability on the results. Also, Chapters 10 and 14 in the NUREG discusses the importance of success criteria to the results in general terms.

5. **Comment:** The NUREG should address the criteria used to determine what constitutes core damage. Many IPEs use core uncovery while others use a peak cladding temperature of 2200°F. This is important in that it impacts what equipment can be used to avoid core damage. (Reference: see Table C.1, #11, 15)

Response:

The impact of the definition of core damage on success criteria is discussed in general terms in Chapters 10 and 14. Specific impacts on the variability of the reported CDF definitions were not addressed because insufficient information was provided in the IPE submittals.

6. **Comment:** A discussion on how the component failure rates and the common cause failure rates impact the results is missing from the NUREG. This could be particularly important for assessing the importance of station blackout (SBO) since the reliability of on-site emergency AC power is critical. (Reference: see Table C.1, #16)

Response:

Because of the variability in the IPE modeling, it is not possible to always ascertain the impact of component failure rates and common cause failure rates. However, these factors were considered in establishing the parameters affecting the variability in the reported CDFs. Selected comparisons were made and, as discussed in Chapter 11, these failure rates were found to be important to the CDF variability. Also, based on a limited survey of data, Chapter 14 indicates that a wide variety of failure rates were identified in the IPEs for some components. This variability applies not only to plant-specific data but also to generic failure rates identified in the submittals.

7. **Comment:** Care must be taken when comparing the CDFs from transient events and from loss of coolant accidents (LOCAs). The IPEs approach the modeling of consequential LOCAs (e.g., reactor coolant pump, RCP, seal LOCAs or stuck-open power-operated relief valves or safety relief valves, SRVs) differently. Sometimes the CDFs from these events are reported in the transient contribution and sometimes in the small or medium LOCA CDF. It needs to be clearly stated how this is handled in NUREG-1560. (Reference: see Table C.1, #16)

Response:

It is true that there was considerable variability among the IPEs with regard to grouping sequences (for reporting) involving consequential LOCAs. However, the majority of the submittals reported sequences initiated by either a rupture or an inadvertent open SRV as LOCAs, and sequences with consequential LOCAS occurring after some other initiator as transients. This format was chosen for categorizing and reporting the results. For those IPEs that did not provide the results according to this format, an attempt was made to regroup the results to allow for comparison with the CDFs for other plants. However, in some cases, insufficient information

was provided in the IPE submittal to distinguish the CDFs associated with these different accident sequences. In those instances, the licensee's reported results for transients and LOCAs were used directly.

8. **Comment:** It is not clear where special initiators fit into the CDF information reported in the NUREG. Generally loss of component cooling water and loss of service water can be important contributors to the CDF for PWRs due to the potential for an RCP seal LOCA. It would be advantageous to report the transient results in terms of CDF due to loss of decay heat removal and the CDF due to consequential LOCAs. (Reference: see Table C.1, #16)

Response:

It is agreed that it would be useful to separate the contributions from loss of decay heat removal and consequential LOCAs for the transient sequences. However, this information is not available consistently from the IPE submittals. Estimates were made from the reported information, whenever possible, and used in the report to identify relevant insights. The NUREG identifies that consequential LOCAs are important contributors to the CDF for many BWRs and PWRs.

9. **Comment:** The discussion on LOCAs should be directed at the ability of plants to mitigate small LOCAs. Overall, large LOCAs are not significant contributors to CDF. (Reference: see Table C.1, #16)

Response:

Significant contributions were observed from different sizes of LOCAs in different submittals. Therefore, it is not always true that large LOCAs are not significant contributors and that the discussion should focus on only small LOCAs. The NUREG discussion identifies what sizes of LOCAs dominate the LOCA contributions in each plant group and the reasons why.

10. **Comment:** A basis for the key perspective that PWRs with better feed-and-bleed capability generally have lower CDFs should be provided. There are many other plant design features and modeling methods that have a greater impact on CDF. (Reference: see Table C.1, #16)

Response:

The observation is made in the context of all PWRs. Within the Westinghouse plant groups, other factors besides feed-and-bleed capabilities are more important for explaining differences in transient CDFs and are discussed in the report. However, differences in feed-and-bleed capabilities are important when comparing across all PWRs because of the Babcock & Wilcox and Combustion Engineering plant design differences.

11. **Comment:** It is not clear from the information presented that the Westinghouse RCP seal LOCA model provides a lower contribution to CDF than the IPEs that used the NUREG-1150 model.

Since this is very important to many plants, it is recommended that NUREG-1560 provide a detailed comparison of the two approaches. One of the dominating factors in the seal LOCA model is the probability of core uncovery occurring within the first hour. IPEs using the Westinghouse RCP seal LOCA model typically use 0.0283 and the NUREG-1150 model uses 0.0. The NUREG-1150 model does not consider any seal leakage for the first 90 minutes. From these facts it appears that the Westinghouse RCP seal LOCA model is more conservative. (Reference: see Table C.1, #16)

Response:

A comparison of the seal LOCA probabilities from the two models was not possible due to the unavailability of the reports documenting the Westinghouse model (with and without seal binding and popping open included). However, the Point Beach IPE and the response to questions concerning the Farley IPE did provide

an opportunity to compare the core uncovery probability as a function of time for cases involving RCPs equipped with the old o-ring elastomer with the vessel either depressurized or not depressurized, and with the RCPs tripped. A comparison of the values from these curve fits with the core uncovery times calculated for identical cases for the Surry plant, as reported in NUREG-1150, Volume 3, is provided in Figure C.1. The curve fit is only valid over the time frame of 30 minutes to 8 hours. It should be noted that Point Beach is a two loop plant while Surry and Farley are three loop plants and thus the core uncovery time for a given leak rate could be different. However, since the reactor coolant system volumes for the plants are roughly scaled by the number of coolant loops, the core uncovery times for three plants for the same amount of leakage from each pump should not be substantially different. Thus, the core uncovery probability comparison in Figures C.1 provides a reasonable picture of the differences between the NUREG-1150 and Westinghouse seal LOCA models.

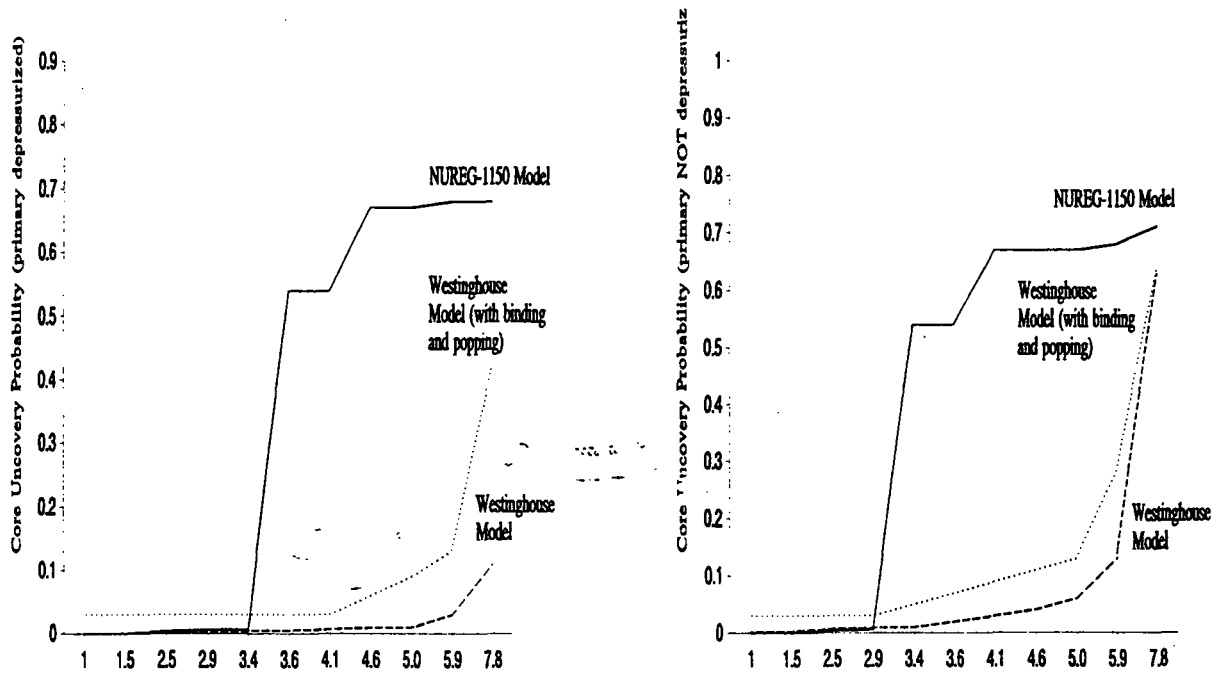

Figure C.1 Comparison of NUREG-1150 and Westinghouse seal LOCA models — old o-ring elastomer.

Figure C.1 indicates that all three models predict small probabilities of leaks and core uncovery for early times (less than about 3 hours). Because of this, differences between the three models do not have a significant impact on CDF for this early time period. However, for later times, the differences are more significant. The Westinghouse models generally predict much smaller probabilities for core uncovery for time periods greater than approximately 3 to 3.5 hours, particularly for cases where the vessel is depressurized. For scenarios where the vessel is not depressurized, however, the probabilities predicted by the Westinghouse models rise sharply at about 8 hours, so that the three models give similar probabilities at that time.

The fact that seal LOCAs occur in all three models does not mean that the impact on the CDF will be the same in both cases. As noted earlier, none of the models result in a significant contribution to CDF in the first three hours. However, unlike the Westinghouse models, the NUREG-1150 model can result in significant contributions to CDF based on core uncovery in the 3 - 8 hour time frame. For example, in this time frame during a station blackout, the core is likely being cooled by auxiliary feedwater, given that battery power is still available. Therefore, without a seal LOCA, core damage would not be expected during this time frame. For times past 8 hours, all three models predict a high probability of a seal LOCA leading to core uncovery. However, for these longer times, battery depletion would have occurred at most Westinghouse plants, leading to loss of heat removal and boiloff. Therefore, if AC power recovery does not occur, core damage will result whether or not a seal LOCA is present. In this situation, the station blackout CDF is not affected by small seal LOCAs that would result in core uncovery at times greater than 8 hours. The precise impact of the model differences is plant-specific, depending on battery depletion times and AC power recovery alternatives. Similar impacts occur for non-station blackout scenarios (e.g., loss of component cooling water events) where the seal leakage rate impacts the time available for other recovery actions such as arranging alternate charging pump cooling.

The documented NUREG-1150 seal LOCA model indicates no seal failure prior to 90 minutes. However, after most of the IPEs were completed, an error in the NUREG-1150 model was identified which indicates that there should be some probability of seal failure immediately after loss of seal cooling. Thus, the contribution of RCP seal LOCAs in the IPEs that utilized the NUREG-1150 model is likely underestimated. An evaluation for the NUREG-1150 study for the Sequoyah plant indicates that the seal LOCA contribution was underestimated by 18% (corresponds to a core damage frequency of 7.7E-7/yr). Accounting for this error would slightly widen the difference between the Westinghouse and NUREG-1150 models.

12. **Comment:** The NUREG discusses uncertainty associated with the Byron-Jackson N-9000 seals and *"infers that the IPEs* [for plants with these pumps] *are suspect in their RCP seal LOCA conclusions."* Details concerning this technical issue have been provided to the NRC in various forms in the past. Please modify the NUREG to reflect the technical information provided and remove the inference that the IPEs are suspect in their RCP seal LOCA conclusions. (Reference: see Table C.1, #8)

Response:

NUREG-1560 reflects the information provided in the IPE submittals which indicate that the contribution from RCP seal LOCAs is generally small for plant with Byron-Jackson pumps. The NUREG reiterates the statements made in the submittals that there is little or no potential for seal LOCAs in these plants if the RCPs are tripped. The submittals cite the design of the pumps some limited analyses, test, and actual experience as the basis for this argument providing some references. No judgement is made in NUREG-1560 about the accuracy of the RCP seal LOCA modeling for these plants based on the information in the submittal. However, the potential for RCP seal LOCAs in these plants is still being reviewed as part of NRC's Generic Safety Issue 23. The information cited in the submittals as the basis for the RCP seal LOCA modeling is being examined as part of the resolution of this issue.

13. **Comment:** The reported CDFs have been rounded to one significant figure. The NUREG should report the actual CDFs reported in the IPEs. (Reference: see Table C.1, #8)

Response:

A decision was made to report the CDFs to one significant figure (to provide consistency) and are based on the actual values reported in the IPE submittals.

C.4 Chapters 4 and 12: IPE Results Perspectives: Containment Performance

1. **Comment:** Conditional containment failure probability (CCFP) is not a good measure of safety performance. The use of conditional measures implies an independence between the systems which prevent core damage and the systems which prevent containment failure which is part of the design of the current generation of light water reactors. Plants with relatively higher CCFPs are not necessarily less safe than those with relatively lower CCFPs. The measures which impact public safety are related to the frequency of releases from the containment. (Reference: see Table C.1, #11, 12, 22, 23)

Response:

One of the main objectives of the chapters in NUREG-1560 related to containment performance is to obtain perspectives on the performance of the various containment types independent from other plant features. For this purpose, the CCFP is a useful parameter since it decouples containment failure from core damage frequency. This was also recognized by the majority of licensees since CCFPs are reported directly in most of the IPE submittals. Ideally, the comparison of containment performance among different IPEs would be accomplished by comparing CCFPs for individual plant damage states. However, such a comparison is not possible since the definition of the plant damage states was left to the individual analyst and thus varies from IPE to IPE. NUREG-1560 also recognizes that the probability of containment bypass is not a measure of containment performance in the same way that isolation or structural failure of the containment is. Therefore, the NUREG separates the conditional probabilities of containment bypass and containment "failure" when making comparisons. The importance of containment failure frequency is acknowledged in Chapter 12 of the NUREG where comparisons of containment failure frequencies as well as release frequencies are also presented. The NUREG does not draw conclusions or make implications regarding overall plant safety based on CCFPs. Containment failure probabilities are used only to compare the containment performance among plants with the same type of containment and among different containment types. For this purpose the CCFP is the best suited parameter.

2. **Comment:** The report utilizes at least five different figures of merit in characterizing containment performance. It is never clear which figure is most appropriate or why. The figures include: total conditional containment failure probability, conditional probability of various containment release types (bypass, early failure, late failure), frequency of bypass and early release, conditional probability of "significant early release," and frequency of releases with the potential to cause early fatalities. (Reference: see Table C.1, #22)

Response:

NUREG-1560 uses various parameters related to containment performance in different chapters of the report depending on the purpose of the comparisons to be made and the perspectives to be obtained. There is no single "most appropriate" containment performance figure of merit for the whole report, nor should there be. Those parameters which best served to illustrate the points to be made for the issues at hand were chosen in different sections of the report. Total conditional containment failure probability is not used in the NUREG. For purposes of obtaining perspectives on containment performance, conditional probabilities of containment bypass,

the CCFPs for early and late failure are used in Chapters 4 and 12. Conditional probabilities of significant or large early release, defined as early releases where releases of Cs, I and Te exceeded 0.1 of core inventory, are also compared in these chapters since this type of release was singled out in many IPE submittals. Finally, frequencies of early release from bypass and early containment failure were used in Chapters 7 and 16 since this parameter was the one which allowed an indirect comparison of the IPE results with the safety goals.

3. **Comment:** While there have been some mis-applications of MAAP, any implication that the MAAP code is inadequate is wrong. It is misleading to state that MAAP does not have a comprehensive treatment of severe accident phenomena. A more problematic item involves the utilities which did not properly apply MAAP and/or relied on the industry position papers. (Reference: see Table C.1, #8, 11, 12, 22)

Response:

MAAP as well as other system level codes do not cover the range of postulated severe accident phenomena (e.g., steam explosions, direct containment heating, shell melt-through, hydrogen detonation). This is what is meant by the statement that the MAAP code does not have a comprehensive treatment of severe accident phenomena. The EPRI report on MAAP acknowledges *"one should recognize that MAAP cannot and does not contain detailed models for all phenomena."* As noted above, other system level codes share this limitation, and this is one reason why the IPE guidance called for proper sensitivity studies to be conducted as part of the Level 2 analysis. In some cases MAAP was applied by the IPE analysts in a way that did not follow industry recommended guidelines. NRC noted *"...the adequacy of the MAAP 3.0B code for use in the IPEs..."* but also stated that *"licensees...bear the burden of proof that they have applied the code properly, and that they meet the intent of the IPE generic letter."* Regarding the industry position papers, their application in an IPE to qualitatively dismiss a number of accident progression phenomena, without any sensitivity considerations, or without any understanding of the uncertainty associated with the different phenomena, is not in line with the intent of Generic Letter 88-20. This approach was less helpful in fostering a licensee's appreciation and understanding of severe accident behavior than a proper application of MAAP.

4. **Comment:** Results are presented by reactor and containment type and NSSS. It would be valuable to also look at the architect/engineer or builder to explain the variation in reported results. (Reference: see Table C.1, #16)

Response:

Early in the IPE Insights Program a decision was made to group the containment performance results under the five common containment classes used in the United States. Containment response to severe accidents has been found to correlate to these five containment classes as illustrated in the NRC's Containment Performance Improvement program. In discussing containment performance perspectives, NUREG-1560 identifies those architect/engineer specific containment construction features which play a significant role in the IPE analysis, as reported in the IPE submittals. These features include the containment material, layout of reactor cavity, and location of sumps and drain lines.

5. **Comment:** It is judged that there is no basis in NUREG-1560 to assert that the observed variability in the IPE results is anything but dominated by plant differences in design, procedure, and training. (Reference: see Table C.1, #15)

Response:

In discussing containment performance perspectives, NUREG-1560 identifies the plant

specific differences described in the IPE submittals which lead to some of the variability in the reported results. However, it is clear that modeling assumptions also play an important role in the observed variability in containment performance. Assumptions regarding the amount and composition of core material exiting from the reactor vessel, the coolability of this debris, and the pressure and temperature rise in the containment due to core debris dispersal are examples of modeling assumptions which had a significant influence on the assessment of containment performance. Other assumptions include the likelihood of in-vessel recovery of the accident, including the likelihood of retaining the core debris in the reactor vessel via external cooling of the vessel.

6. **Comment:** It would seem prudent to avoid misinterpretations by providing the specific NRC assumptions used in extrapolating IPE submitted words to the construction of the comparisons among plant results in NUREG-1560. These assumptions would include:

- What the relationship of containment vent treatment is to the CCFP, the early releases, and other measures of risk;
- what the correlation is between each IPE result for early and late releases and their definition of "early" and "late";
- how the assignment of multiple containment failure modes affects the assignment of the allocation of failure modes in comparisons (e.g., shell melt-through following wetwell failure); and
- defining the treatment of dynamic failure modes and their associated failure locations as it relates to inferences about failure locations and timing. (Reference: see Table C.1, #15)

Response:

There exists detailed discussion in the appropriate sections of Chapters 4 and 12 of NUREG-1560 on:

- how venting was grouped to the different containment failure modes.
- how "early" and "late" was defined in the comparison of failure modes and releases.
- how multiple containment failure modes were treated as they were reported in the IPE submittal (i.e. whichever failure mode was considered dominant in the submittal base case results was the one used in NUREG-1560).
- The above comment on the treatment of dynamic failure modes is not clear, and no further clarification was provided at the workshop; consequently, no changes were made to NUREG-1560.

C.5 Chapters 5 and 13: IPE Results Perspectives: Human Performance

1. **Comment:** It is stated in the report that in most cases there is little evidence that the human reliability analysis (HRA) quantification method per se has a major impact on the results. This seems to imply that *"the impact of HRA on PSA can best be described as indeterminate"* or *"that the HRA seems to have little effect on the results of the PRA."* If this is the case, why are the HRAs identified as important shortcomings of the IPEs and why is the quality of the HRAs a concern. (Reference: see Table C.1, #8, 11, 12, 15, 21)

Response:

The interpretation that *"the impact of HRA on PSA can best be described as indeterminate"* or

that *"the HRA seems to have little effect on the results of the PRA"* is not what was meant. How and how well the HRA method is applied and the resulting human error probabilities (HEPs) clearly have significant impacts on the results of the PRA. Thus, it is for this reason that concern is raised in the NUREG about the "quality" of the HRAs performed by the different licensees. The statement that *"in most cases there is little evidence that the HRA quantification method per se has a major impact on the results,"* was meant to imply that the HRA results from the different IPEs did not in general appear to vary directly as a function of the particular or "nominal" HRA method used, e.g., the Technique for Human Error Rate Prediction versus the Success Likelihood Index Methodology versus the Human Cognitive Reliability model. The variability in results appeared to be more a function of how or how well the HRA methods were applied or the impact of plant-specific characteristics, as opposed to which nominal HRA method was used. Due the confusion caused by the statement and the fact that the direct impact of the nominal method per se is difficult to evaluate given the many other relevant factors, the statement was deleted from the final NUREG. Additional clarification regarding the quality of the HRAs performed for the IPEs is provided below in the response to Comment #2.

2. **Comment:** In spite of the assertion in the report that *"it appears that there are reasonable explanations for much of the variability in HEPs and in the results of the HRAs across the different IPEs,"* it is also asserted that because *"many of the licensees failed to perform high-quality HRAs, it is possible that the licensees obtained HEP values that are not appropriate for their plants."* These statements appear to be inconsistent. Moreover, others sections of the report indicate that not all of the variability in HEPs could be explained. Please provide clarification on what appears to be inconsistent statements and address the assertion that *"many of the licensees failed to perform high-quality HRAs."* (Reference: see Table C.1, #1, 2, 8, 11, 12, 22)

Response:

Confusion arose regarding the implication or meaning of the significant variability in HEPs that was identified for selected human actions across plants, particularly in terms of the quality of the HRAs. Figures displaying the HEPs for several events (e.g., manual depressurization during transients) were presented in the report and discussions of the reasons for the variability were provided. Many of the comments received from licensees on this topic attempted to defend the variability on the basis of the numerous reasonable factors that would lead to the variability. That is, the values across plants may have been developed on entirely different bases. For example, different plants have different system characteristics and may have different procedures. Initiator and sequence-specific factors and dependencies will also lead to variability in HEPs. Moreover, some plants only used "screening values" in modeling some of the examined events. On the basis of these and other factors, the commentors indicated that such variability would be expected.

This conclusion is, at least in part, one point the staff was trying to make and which was stated in the report. That is, there are *"reasonable explanations for much of the observed variability in HEPs across plants."* In other words, the rather striking degree of variability, in at least nominally similar human actions, is based to some extent on valid differences. From this perspective it can be argued that the licensees attempted to consider relevant factors in obtaining the HEPs for operator actions and that the results of the HRAs performed by the different licensees were generally consistent and therefore useful. In fact, the staff does not in general disagree with this conclusion.

However, another conclusion reached by the staff and documented in this report was that not <u>all</u> of

the variability in the examined HEPs was easily explained. That is, after "acceptable" reasons for variation were considered, there still appeared to be some degree of unexplained variation the HEPs (see Chapter 13). While some of this variation would be expected due to the lack of precision in existing HRA methods, it is also possible that some of the variation was due to factors such as analyst biases, invalid HRA assumptions made by analysts performing the HRAs, or superficial HRA analyses that failed to adequately examine and model the potential for human error (e.g., through careful consideration of plant-specific performance shaping factors (PSFs), consideration of dependencies, use of simulator exercises, etc). Due to the limited information provided in many submittals on the derivation of particular HEPs, it is difficult to determine the extent to which inappropriate factors actually influenced the derived HEPs. However, examinations of the submittals during the project indicated that not all licensees performed quality HRAs. That is, not all licensees applied the existing HRA methods as well as they could have. For example, they did not always consider dependencies, appropriately assess the impact of time availability, or carefully consider plant-specific PSFs. Some failed to model pre-initiator actions and others did not conduct simulator exercises or perform walkdowns and timing of operator actions to be conducted outside the control room, etc. The conclusion that not all licensees conducted high-quality HRAs is further documented in some of the staff evaluation reports (SERs) that have been issued on the submittals. Some submittals indicated as having met the intent of Generic Letter 88-20 were found to have various weaknesses that could have influenced the HEPs obtained for particular events.

While the degree of consistency in HEPs obtained for similar human actions in similar contexts suggests that in general the HRA results from the IPEs were useful in terms of meeting the intent of Generic Letter 88-20, it should be further noted that even when reasonable consistency exists, it is not necessarily the case that all the HEPs calculated by a particular plant were realistic and valid for that plant. As noted in Chapters 5 and 13, reasonable consistency can be obtained in HRA without necessarily producing valid HEPs. An HEP is only valid to the extent that a correct and thorough application of HRA principles has occurred. For example, if a licensee simply assumed (without adequate analysis) that their plant is "average" in terms of many of the relevant PSFs for a given event, but then does appropriately consider the time available for the event in a given context, the value obtained may be similar to those obtained for other plants with similar time frames for the event. Yet, the resulting value may be optimistic or pessimistic relative to the value that would have been obtained if the licensee had conducted a detailed examination of the relevant plant-specific factors. Thus, while the degree of consistency obtained by the licensees is encouraging regarding the ability to compare the results of the IPEs, and while many licensees performed excellent HRAs, the fact that some licensees did not perform as thorough HRAs as possible given the state-of-the-art in HRA at the time, means that the results are not as good as they might have been. It does not mean that individual licensees and the industry in general did not obtain important information from performing the IPEs.

3. **Comment:** By questioning the quality of the HRAs performed for the IPEs, NUREG-1560 seems to imply that the licensees should have attempted to extend the state-of-the-art in HRA in order to obtain quality results. (Reference: see Table C.1, #8, 11, 21)

Response:

The staff believes that the state-of-the-art in HRA at the time of the IPEs was adequate for the intent of Generic Letter 88-20. The shortcomings related to the HRAs performed for the IPEs were in how the existing methods were applied, rather than the methods themselves. Of course, this

position does not imply that improvements are not needed in HRA, but rather that useful results can be obtained with thoughtful and thorough applications of existing methods.

4. **Comment:** The NRC needs to initiate a number of policy and research activities to address shortcomings both in the NRC's attitudes and strategies for ensuring that the licensees maintain safe plants and in the development and use of PRA and HRA methods and techniques. These activities (summarized) include establishing a regulatory attitude that encourages the licensees to be pro-active rather than reactive (to the NRC) in ensuring plant safety, encouraging more thorough and realistic HRAs, supporting the development of multiple new approaches to HRA (which include more effective use of simulators), reevaluation of the real contribution of common cause to risk, reevaluating the use of Bayesian updating during "period of rapid changes in maintenance," and investigating the impact of management and organizational factors on plant safety. (Reference: see Table C.1, #4)

Response:

The author (of the comments summarized above) acknowledged that the *"comments are not just on the NUREG document itself, but are also directed towards some overall aspects of PRAs and HRAs."* However, none of the comments appear to address the NUREG itself. Nevertheless the NRC does currently have programs addressing each of the issues raised by the author, e.g., development of improved HRA methods and consideration of the impact of management and organizational factors on plant safety. Further, the NRC staff has reviewed the comments and will consider them in future directions of research.

C.6 Chapters 6 and 14: IPE Models and Methods Perspectives

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Numerous comments were received on the description of a "quality" PRA in Chapters 6 and 14 and on the comparison of the IPEs to a quality PRA in Chapters 6 and 15 of the draft NUREG. Several commentors felt that these chapters were inappropriate for NUREG-1560 and that they should be deleted from the final report. This recommendation was largely driven by the assumption that the attributes of a "quality" PRA were intended to be standards or requirements and that all the attributes had to be met prior to using PRAs in future risk-informed regulatory activities. Given that some commentors felt that the PRA attributes were too demanding, overly prescriptive and beyond the current state-of-the-art, it follows that if they were assumed to be requirements then they could be interpreted as a significant burden on the industry. Several comments emphasized that the scope and attributes of a PRA to be used for risk-informed regulatory activities should be commensurate with the application. This implies that PRAs with significantly less attributes and of more limited scope than the PRA described in NUREG-1560 would be acceptable for risk-informed applications. Other commentors stressed that any applications of the PRA attributes in NUREG-1560 to the creation of an industry standard should be viewed as developmental in nature. An industry-wide standard for PRA quality should be based on a broader and more deliberate development effort that involves practitioners from various

organizations. (Reference: see Table C.1, #1, 2, 8, 9, 15, 16, 20, 22 and 26)

Response:

Chapters 6, 14 and 15 of draft NUREG-1560 have been significantly revised for the final report. Specifically, Chapters 14 and 15 have been replaced with a new Chapter 14, and references to the use of the IPEs in risk-informed regulation have been removed. Chapters 6 and 14 in the final report summarize PRA characteristics and state that they:

- are not "standards" nor do they represent regulatory guidance.
- are included only as a benchmark in order to draw perspectives on the models and methods used in the IPEs.
- do not define the needed quality or scope of the PRA elements needed for a particular regulatory application.

2. **Comment:** Several comments were related to the following statement in draft NUREG-1560, *"...and other utility personnel are excluded from the peer review team."* This statement was interpreted by some commentors as implying that no employees of any utility can serve as a peer reviewer. (Reference: see Table C.1, #1, 8, 12, 15, 16, 20 and 22)

Response:

This interpretation was not intended. The statement was included simply to indicate that it would be inappropriate for utility staff to be part of the PRA peer review team for plants owned and operated by their utility. NUREG-1560 has been revised accordingly.

C.7 Section 7.1 and Chapter 15: Safety Goal Implications

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comments and associated responses are provided below.

1. **Comment:** The concern is that the results reported in the original IPE submittals are not current and could be misleading when compared to the Safety Goals. For example, several plants identified in Chapters 7 and 16 in Draft NUREG-1560 (Chapter 15 in Final NUREG-1560) as potentially approaching the early fatality quantitative health objective (QHO) have subsequently updated their PRAs with significant reductions in CDF and large early release frequency, LERF (including Browns Ferry, Beaver Valley and Palo Verde). (Reference: see Table C.1, #22, 25 and 26)

Response:

NUREG-1560 has been revised to clarify that the perspectives on the safety goal are based on the original IPEs/PRAs which may have subsequently changed. However, the results quoted in NUREG-1560 will not be revised. New information obtained by the staff will be included in NUREG-1560 (see Appendix B). In the case of the safety goal comparisons if any of the plants that were identified as approaching the early fatality QHO submit revised results, this will be noted in Chapter 7 and 15 and the reader will be directed to the appendix.

2. **Comment:** Inferences that a few plants may approach the early fatality health objective based on a comparison of the IPE and NUREG-1150 results may not be valid. Additional insights gained from the containment performance evaluations and recent research in the area may

lead to different conclusions than the NUREG-1150 analyses. (Reference: see Table C.1, #16)

Response:

NUREG-1560 has been revised to clarify that NUREG-1150 containment results were not used to link the IPE results to the safety goals. For early fatality risk, a two step process was used. In the first step, the frequencies of early containment failure and bypass were obtained from the IPEs and plants with low frequencies ($<10^{-5}$/ry) were screened out from further consideration. For the remaining plants, the frequencies of source terms with relatively large release fractions (>0.03 Cs, I, Te) were obtained. The source term frequencies were then adjusted for population and compared to the goal.

3. **Comment:** There is an implication in the report that the only way a comparison can be made to the "Safety Goal" is to have a Level 3 PRA. Such a PRA was never mandated, requested or suggested by the NRC and there are a number of ways to compare to the Safety Goal other than having a Level 3 PRA. The NUREG could address how the NRC and industry (there are several EPRI documents and other papers, positions and reports) have defined or linked the NRC "Safety Goal" in terms of Level 1 and 2 surrogate indicators. (Reference: see Table C.1, #15, 21)

Response:

The approach used by the staff in Chapters 7 and 15 of NUREG-1560 was based on using Level 1 and 2 surrogate indicators to link the IPE results to the safety goals. The wording in Section 6.4 has, therefore, been changed to make it clear that a Level 3 PRA is not the only way to make a comparison to the safety Goals.

4. **Comment:** One comment stated that conclusions based on using the IPE results for comparisons to the QHOs of the safety goals must be carefully qualified. The purpose of the IPEs was not to define absolute risk levels, but rather to identify plant severe accident vulnerabilities. Consequently, the safety goal computations performed by the staff (described in Chapters 7 and 16 of draft NUREG-1560) are not an adequate technical basis on which such a conclusion can be drawn. In a related comment, SECY-90-104 was quoted, *"based on the significant additional resources that would be required to make a meaningful comparison of the IPE results with the safety goal policy statement and the potential problems associated with using the as-submitted IPE data, the staff recommends that no direct comparisons be made unless the IPEs are reviewed to a greater level of detail than currently planned."* As the commentor believes that a review of greater detail did not occur, it was recommended that the direct comparison of IPEs to the Safety Goals in Chapters 7 and 16 be removed from the final NUREG. (Reference: see Table C.1, #19, 22)

Response:

The final version of NUREG-1560 has been revised to clearly describe the limitations of the approach used to compare the IPE results to the safety goals and subsidiary objectives. However, the use of Level 1 and 2 indicator (CDF and LERF) as surrogates for the safety goals is consistent with recent industry positions (refer to Comment #3 in Section C.7 above) and consistent with the guidance provided by the NRC for use of PRAs in risk-informed regulatory applications (Ref. C.5). The manner in which the IPE results are compared to the safety goals is consistent with the "Integration Plan for Closure of Severe Accident Issues," SECY-88-147 and also consistent with the recommendations of SECY-90-104, namely, *"....indirect comparison of the IPEs and other available PRAs with the Safety Goals, focusing on the insights gained and the adequacy of regulations, is planned."* The SECY further recommends that the *"staff evaluate the IPE results as a whole and summarize any conclusions and recommendations for the*

Commission at the completion of the IPE process."

5. **Comment:** Several verbal and written questions were received at the workshop related to the appropriateness of the current safety goals and the manner in which comparisons were made to these goals. (Reference: see Table C.1, #23, 26)

Response:

The appropriateness of the current safety goals is a policy issue and outside the scope of NUREG-1560. The use of Level 1 and 2 indicators as surrogates for the safety goals is consistent with the staff's guidance provided in the recently published regulatory guides (Ref. C.x).

6. **Comment:** The definition of an early release, particularly a large early release, and the time available for effective evacuation after declaration of a general emergency appears to be arbitrary. Consideration of the accident timing, the site, and the impacts on evacuation (such as an SBO) need to be considered. (Reference: see Table C.1, #25)

Response:

A unique definition of a large early release was not provided in NUREG-1560. A large early release is defined in the staff's regulatory guides (Ref. C.x) on the use of PRA in risk-informed regulation. Numerical objectives for the frequency of a large early release are also provided in those documents. The frequencies of early containment failure and bypass were used in NUREG-1560 to screen out plants with low frequencies. The frequency of source terms with relatively large release fractions were then examined in more detail to estimate the potential early health effects. The assumption was made that these releases occur prior to effective offsite evacuation. This assumption could overestimate the potential for early health effects.

C.8 Section 7.2 and Chapter 16: Impact of Station Blackout Rule on Core Damage Frequencies

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Evaluation of the SBO rule would benefit from a review of the results by Architect/Engineer and not just by reactor type. (Reference: see Table C.1, #16)

   Response: As is discussed in the response to similar comments on Chapters 3 and 11 (and the report in general), early in the IPE Insights Program the plants were grouped by architect/engineer and the IPE CDFs within and among these groups were compared. No strong correlation with the architect/engineer was found because there is considerable design variability even among plants designed by the same architect/engineer. A decision was made to perform the analysis using plant groups based upon the NSSS vendor to account for basic NSSS design differences. The BWRs were further sub-categorized by vintage to account for differences in ECCS design. The Westinghouse plants were grouped according to the number of loops since the ECCS and other general plant features for the plants in each of these groups are generally the same (see Table 10.3). It is recognized that the balance of plant including support systems for plants in each of the designated groups can be different and skew any comparison of the results for a plant group. The NUREG consistently identifies that these plant-specific features impact the results and draws the appropriate conclusions on the resulting insights. Finally, it is recognized that further subcategorization of plants according to a selected parameter could be made. However,

variability in other parameters would likely impact that comparison. Because of this fact and also due to resource limitations, further subcategorization was not pursued.

C.9 Section 7.3 and Chapter 17: Comparison with NUREG-1150 Perspectives

Several comments were received expressing technical disagreement with some of the information provided in these chapters. The text was revised where appropriate. In addition, several general comments were provided on the content of these chapters. These comment and associated responses are provided below.

1. **Comment:** Chapter 18 in Draft NUREG-1560 (Chapter 17 in Final NUREG-1560) presents a comparison of NUREG-1150 results with IPE results as a whole. A more interesting comparison would be between the individual NUREG-1150 results and the corresponding IPEs. This would provide a more detailed information on specific modeling issues. (Reference: see Table C.1, #2)

   Response:

   Section 7.3 indicates that the focus of NUREG-1560 is on comparing global perspectives discussed in NUREG-1150 with the overall results of the IPEs. A plant-specific comparison between NUREG-1150 and the applicable IPE analyses are provided in the individual SERs on the five IPEs. Chapter 17 in the Final NUREG-1560 has been revised to clarify the scope of the comparison in NUREG-1560 and to note that plant-specific comparisons may be found in the SERs.

C.10 Chapter 8: Overall Conclusions and Observations

Some general comments concerning the content of Chapter 8 were received from several organizations and individuals. Responses to these comments are provided below.

1. **Comment:** Due to the nature of the IPE process requested in Generic Letter 88-20 (a search for vulnerabilities, not characterization of absolute risk), the applicability of the IPE results for regulatory follow up activity should be limited. Section 8.2.4 states that the NRC staff plans follow-up activities to determine if additional regulatory actions are warranted for plants with relatively high CDFs or CCFPs. NUREG-1560 does not consider revised CDF and CCFP values provided to the NRC, which in some cases, are substantially different than the original IPE submittal values. Consequently, use of the IPEs for comparison to safety goals, identification of "outlier" plants, and for direction of inspection and follow-up activities should be minimized. Such actions have the potential to lead to ineffective use of NRC staff and utility resources in pursuing areas which are known to be outdated. The NRC staff should evaluate these changes in the plant CDF and CCFP values before planning follow-up activities. (Reference: see Table C.1, #20, 22)

   Response:

   The IPE results and insights provide a useful source of information for identifying areas where follow-up activities might be warranted. The information contained in NUREG-1560, however, is merely a starting point and is by no means the sole basis for regulatory decisions. Before any plant-specific actions are taken, the best available information will be considered, including any revisions to the original IPE submittals, recognizing that most of the newer information has not yet received staff review. Further, any

proposed regulatory actions are subject to the Backfit Rule as described in 10CFR50.109.

2. **Comment:** The NRC staff's approach in looking at CDF and CCFP as independent factors is incorrect. It assumes the existence of either a high CDF or high CCFP is evidence on its own of a potential concern. In reality, the two factors should be looked at together. They are each a part of the overall input to risk, which should be the figure of merit (the CDF/CCFP criteria do not have any established technical connection to the QHOs of the Safety Goal). (Reference: see Table C.1, #8, 22)

Response:

The major objectives of the IPE Insights Program are outlined in both the Forward and Introduction of NUREG-1560. For at least one of those objectives (i.e., providing perspectives on plant feature and assumptions that play a role in the estimation of CDF, containment performance and human performance), it is useful to look at CDF and CCFP separately. The use of these parameters in NUREG-1560 does not imply that a high value for either parameter alone is a potential concern or will be the basis for regulatory decisions. Instead, the use of these parameters allows the staff to focus individually on the Level 1 and Level 2 analyses performed for the IPEs, thereby accomplishing the objectives noted above.

3. **Comment:** Concerning any follow-up regulatory activities, it's suggested that the investigation and regulatory considerations not be limited just to the high CDF or CCFP issues. Areas where the risk impact is small and the safety benefit is not appreciable should also be investigated for reduced regulatory burden. (Reference: see Table C.1, #6, 16)

Response:

The primary focus of the NRC is to assure the safety of the public. Therefore, it is natural that the NRC tends to be more concerned with eliminating vulnerabilities and reducing risks than with reducing burden. However, the latter objective is desirable and the NRC encourages the industry to submit requests for reduced regulatory burdens in areas where they believe that risks are low and substantial cost savings can be achieved.

4. **Comment:** The discussion of the Maintenance Rule says it is acceptable to use the IPEs to determine risk significant systems. However, this is not compatible with the findings about the usefulness of the IPEs for risk-informed regulation. Likewise, the NRC implies that for inspection purposes the IPEs are adequate for them to target areas for plant-specific inspections but NUREG-1560 states that the PRAs are only adequate to identify dominant accident sequence types and their relative importance. This seems inconsistent. Furthermore, the NRC seems to be attempting to use PRA information in a selective manner, where it serves their purposes. (Reference: see Table C.1, #22)

Response:

References to the use of the IPEs in risk-informed regulation have been removed from the final version of NUREG-1560. Issues related to the quality and scope of PRAs needed for risk-informed regulation are discussed in the staff regulatory guides, and standard review plans. The role of the IPEs in risk-informed regulation will be determined in the context of these documents, not NUREG-1560.

5. **Comment:** The report implies that until "quality" PRA requirements are fully met, PRAs cannot be used for any regulatory purposes. If that is the case, "as is" PRAs are inappropriate to support such areas as the Maintenance Rule and Technical Specification changes. Such an interpretation is counterproductive and is not supportive of the PRA Policy that looks to enhance use of PRA in regulation commensurate with the state-of-the-art technology. Recognized

weaknesses, and tools to deal with those weaknesses delineated in the Standard Review Plan makes the "as is" PRA applicable for a wide variety of applications while "quality" PRA requirements are phased in. Waiting until perfect "quality" of PRA is achieved before utilizing the results is impractical. It is expected that "quality" and "standardization " will evolve, not through a priori definition, but through frequent, repeated application and peer review of PRAs. (Reference: see Table C.1, #15, 17)

Response:

The comment is similar to comments received on Chapters 6 and 14 (refer to Comment #1 in Section C.6). These chapters and Chapter 8 have been significantly revised for the final report. It was not intended to imply that all the attributes in draft NUREG-1560 have to be met before a PRA can be used to support risk-informed regulatory applications.

6. **Comment:** The NUREG states the NRC staff plans to conduct follow-up activities to monitor implementation of the potential plant improvements identified by the IPEs. The improvements were identified as "potential improvements" which in most cases were identified as areas for further review. The NRC seems to be taking them as having been commitments. These improvements should not be treated as commitments unless the utility clearly identified them as commitments. (Reference: see Table C.1, #8)

Response:

The NRC recognizes that the potential improvements are not commitments in a regulatory sense. However, in many cases the improvements were credited in the IPE. Therefore, if the licensee uses the IPE in future submittals to the NRC, it is important for the NRC staff to know if the credited improvements have been made.

7. **Comment:** The use of NUREG-1560 for a variety of issues is discussed in Chapter 8. However, most of the discussions are actually related to the use of the IPEs to address these issues. NUREG-1560 should not be the source of information for applications as discussed in Chapter 8. The IPEs/PRAs are the primary source and should be used. (Reference: see Table C.1, #8)

Response:

NUREG-1560 summarizes a great deal of important safety information and provides a starting point for identifying and addressing a number of important safety issues. As such, it is an important document and staff resource. However, the staff recognizes that some of the information is out of date and that the individual submittals contain more information. For any particular issue, the staff will use the best available information, including any new submittals, recognizing that some of the new information may require additional review. NUREG-1560 also provides comparisons among the IPEs on selected issues, and this information is useful to the staff when evaluating the treatment of an issue by a particular plant.

C.11 Chapter 10: Background for Obtaining IPE Perspectives

Several comments were received concerning the accuracy of the information provided in this chapter. Corrections were made to the text where appropriate. No general comments were made concerning this chapter.

REFERENCES FOR APPENDIX C

C.1 Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Summary Report, Draft," Vol. 61, No. 221, Page 58429, November 14, 1996.

Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Volume 2, Parts 2-5, Draft," Vol. 61, No. 239, Page 65248, December 11, 1996.

C.2 Federal Register, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance, Volume 2, Parts 2-5, Draft," Vol. 61, No. 239, Page 65248, December 11, 1996.

C.3 NRC Memorandum (From Mary Drouin to M. Wayne Hodges), "Draft NUREG-1560 Public Workshop Summary Report," October 3, 1997.

C.4 ACRS meetings on IPE insights:

November 18, 1993	January 26, 1996 (subcommittee)
December 10, 1993	February 8, 1996
September 27, 1994	May 23, 1996
October 7, 1994	June 11, 1996 (subcommittee)
December 7, 1995	June 12, 1996

C.5 USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Draft Regulatory Guide DG-1061, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," Draft Regulatory Guide DG-1062, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance," Draft Regulatory Guide DG-1064, June 1997.

USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Draft Regulatory Guide DG-1065, June 1997.

Attachment 7

Summary of Results of IPEEE Reviews

On June 28, 1991, the NRC issued Supplement No. 4 to Generic Letter (GL) 88-20, which described the objectives and overall logistics of the Individual Plant Examination of External Events (IPEEE) program, for the evaluation of external events including seismic events, internal fires, and high winds, floods, and other (HFO) external initiators. The primary goal of the IPEEE program has been for licensees to "identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements."

In addition to the principal intent of GL 88-20, the four supporting IPEEE objectives have been for each licensee to:

1. develop an appreciation of severe accident behavior;

2. understand the most likely severe accident sequences that could occur at the licensee's plant under full power operating conditions;

3. gain a qualitative understanding of the overall likelihood of core damage and fission product releases;

4. reduce, if necessary, the overall likelihood of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

The recommended guidelines of NUREG-1407, "Procedural and Submittal Guidance for the IPEEE for Severe Accident Vulnerabilities," and Supplement 5 to GL 88-20 have been developed by the NRC to help ensure that each of these objectives is met.

Preliminary perspectives have been obtained from ongoing technical reviews of 24 IPEEE studies that have been submitted by licenses. These results primarily include: (a) summaries of findings reported in IPEEEs, and (b) perspectives on strengths and weaknesses of licensee submittals in achieving the IPEEE objectives. The IPEEE program appears to have been generally successful in meeting the overall intent of GL 88-20. However, the degree and consistency of such success have varied considerably from study to study, and have been strongly dependent on the level of detail, and methods and assumptions employed in the IPEEE analyses.

Based on the continuing review of the first 24 IPEEE submittals, it appears that the IPEEE program has led to an increase in overall licensee appreciation of severe accident behavior for external events. As requested in NUREG-1407, each IPEEE has involved a seismic evaluation, an analysis of internal fires, and an assessment for HFO events. These evaluations have assessed the potential for externally initiated severe accidents, and have evaluated plant-specific behavior in responding to potential severe accidents.

Consistent with the guidance of NUREG-1407, the emphasis in conducting IPEEEs has been on obtaining a qualitative, as opposed to quantitative, understanding. As expected, therefore, the IPEEEs do not generally convey a definitive ranking of the risk-significance of severe accident sequences or of the dominant risk contributors. Rather, by means of systems modeling and screening analysis, licensees have obtained a greater awareness of severe accident sequences and an improved sense as to the most important among those sequences.

By means of IPEEEs, licensees have been able to generally ascertain whether the risk of core damage associated with each external initiator is comparatively negligible (i.e., falling below the 10^{-6} screening threshold), low, moderate, or high. In some cases, this understanding occurred through direct quantification of core damage frequency (CDF), whereas in other cases, this understanding resulted from having knowledge of the hazard in conjunction with an assessment of the plant's ability to withstand the hazard.

It is important to note that, although in many cases licensees have reported numerical risk estimates (for CDF or frequency of significant radiological releases), the accuracy of such estimates is frequently limited due to simplifying assumptions and approximate procedures employed in the analyses. Hence, the results serve only as general indicators of risk level, and a comparison of CDF results between plants is not particularly meaningful.

Based on the first 24 IPEEE submittals, a majority of licensees have implemented or proposed plant modifications that have a beneficial effect on plant safety with respect to external events. Such plant modifications include hardware changes, procedural changes, and implementation of severe accident management guidelines. Consistent with the qualitative nature of the IPEEE program, it is not usually possible to deduce the numerical risk reductions achieved by these modifications. However, some licensees have employed PRA in their IPEEEs as a means for determining whether or not plant modifications are warranted based on cost-benefit rationale.

Licensees have in most cases followed the guidance in NUREG-1407 in performing their IPEEE assessments. The guidance permits alternative methodologies. For example, there are various approaches for the seismic evaluation, including; seismic margin assessment (SMA) using the NRC methodology, SMA using the Electric Power Research Institute (EPRI) methodology, or seismic PRA methodology. Out of the 24 IPEEE submittals reviewed to date, thirteen have been based on seismic PRA methodology, whereas seven have been performed using the EPRI SMA methodology; one has adopted both the seismic PRA and EPRI SMA approaches; one has been based on the NRC SMA methodology; and two have been performed using a site-specific seismic evaluation approach, in consideration of the "Optional Methodologies" provision of NUREG-1407.

NUREG-1407 has also allowed for the implementation of alternative approaches for the evaluation of internal fires and HFO events. For fire IPEEE evaluations, licensees have implemented EPRI's fire-induced vulnerability evaluation (FIVE) methodology, fire PRA methodology, or a combination of these approaches. For evaluation of HFO events, the licensees either demonstrated that criteria of the NRC's 1975 Standard Review Plan (SRP) were met, or conducted at least one of the following forms of analysis: screening assessment, bounding analysis, or PRA methodology.

These observations highlight an important fundamental difference between the IPE process and the IPEEE process. In the IPE process, comparatively detailed PRA investigation has been invariably implemented, whereas in the IPEEE process, a mix of deterministic methods and detailed PRA investigations has been applied, as well as a mix of screening analyses, simplified hazard-based analyses, and/or bounding PRA-based approaches.

Licensees have performed or proposed various IPEEE-related improvements for their plants in seismic, fire, and/or HFO events. In the seismic area, hardware fixes have included items such as: anchoring equipment, bolting cabinets together, improving existing anchorage or supports, installing missing fastener and bolts, installing spacers on battery racks, eliminating potential interaction concerns, and replacing vulnerable relays. Maintenance actions have included the removal of corrosion on equipment anchorages and application of corrosion protection. Maintenance procedural enhancements have included provisions for proper storage of ladders, tools, gas cylinders, etc., and for proper parking of cranes and chain hoists.

Improvements to fire protection systems include hardware modifications and enhancements to, or development of, fire-response procedures. Additionally, improvements have often taken the form of severe accident management guidelines that address specific accident scenarios related to internal fires, potential effects of wind-induced missiles, and external flooding. Implementation of some of the severe accident management guidelines has led to the acquisition or access of temporary or portable equipment (pumps, diesel oil tanker trucks, etc.). One HFO IPEEE reported the strengthening of the stacks of two adjacent fossil-fuel units to reduce the high-wind risk, and refurbishment of a flood wall to reduce flood risk. The IPEEEs have also, in some cases, referenced plant improvements that had been made (or proposed) prior to the IPEEE, since those improvements resulted in a beneficial effect on plant safety in the face of seismic, fire, and/or HFO events. For example, at one plant, the addition of diesel generators was identified as a plant improvement in the IPE, and was correspondingly reported in the IPEEE since it reduces the risk of station blackout for seismic, fire, and HFO events.

A number of important perspectives and insights have been derived from the NRC's overall review activities pertaining to IPEEE submittals. Some of these key observations are described in the following paragraphs for the seismic, fire and HFO aspects of the IPEEE program. It was stated previously that the intent of GL 88-20 appears to be broadly met by the IPEEE submittals; however, the quality of the submittals has varied considerably from plant to plant. Some of the weakness in the submittals are discussed below. When these weakness have been observed during the review of the submittals, the staff has sent requests for additional information (RAI) to the licensees to complete its assessment of the submittal.

Seismic Events

Key observations obtained from a review of seismic IPEEEs include the following:

- A seismic walkdown was performed for each plant, and in most cases, the walkdown identified conditions pertaining to anchorage, interactions, maintenance, and/or housekeeping that required further investigation. As a result, a number of plant-specific fixes have been implemented at many plants.

- In seismic PRA studies, different hazard curves have been used (i.e., 1993 Lawrence Livermore National Laboratories [LLNL], 1989 LLNL, 1989 EPRI, and individual licensee-sponsored contractor results) from plant to plant. Hence, it is difficult to achieve a meaningful comparison of seismic CDFs across plants. However, the ranking of dominant contributors has consistently been reported in seismic IPEEEs as being insensitive to use of the EPRI or LLNL seismic hazard curves.

- Simplifications in systems analyses, unsubstantiated assumptions regarding human error rates, and use of simplified screening fragilities have, in some cases, obscured findings pertaining to dominant seismic risk contributors and produced unrealistic (high or low) CDF estimates.

- Although the analytical methods vary and there are large uncertainties, it appears that the CDF contribution from seismic events can, in some cases, approach that from internal events.

Fire Events

Key observations obtained from a review of fire IPEEEs include the following:

- No fire vulnerabilities have been reported in the first 24 IPEEE submittals; however, fire-initiated accidents have been found to be an important component of the external events CDF contribution.

- While no plants have identified fire vulnerabilities, about half (of the 24 reviewed) have reported some fire-related plant improvement as a result of the IPEEE effort. These improvements include changes to existing procedures, development of new procedures, or plant modifications.

- Overall licensees have expended a considerable level of effort in conducting fire IPEEEs. A few submittals clearly demonstrated the proper application of fire risk methodologies and data. However several weaknesses have been noted in applying the selected methods and data in some of the fire analyses which affect the robustness and completeness of the submittals. Some of these weaknesses are as follows:

  - Operator actions in response to the effects of fire on systems have rarely been modeled in detail.

  - Several submittals have used questionable methods, procedures, or data for fire damage modeling.

  - Several submittals have used the Nuclear Safety Analysis Center (NSAC)/181 and/or the EPRI Fire PRA Implementation Guide documents for which some optimistic guidelines and data have been identified.

- The possibility of active barrier failure, which may have a significant probability of occurrence, has not been included in most analyses. The significance of active fire barriers is a function of plant layout and separation of redundant trains. Also, the potential for barrier failure associated with large quantities of combustible materials concentrated in one area has not been considered in most of the submittals.

- Although the analytical methods vary and there are large uncertainties, it appears that the CDF contribution from fire events can, in some cases, approach that from internal events.

HFO Events (i.e. high winds, floods, transportation accidents and nearby facility)

Key observations from HFO IPEEE submittals include:

- Transportation and nearby facility accidents have been screened out in all of the 24 IPEEEs that have been reviewed.

- The HFO IPEEE program has had some impact on plant safety. For some plants a greater appreciation of the potential risk impact of high winds/tornadoes and external flooding/dam breaks has resulted from the IPEEE program. Some licensees have implemented or proposed plant improvements with respect to procedural enhancements, severe accident management guidelines, and hardware installation. Procedural enhancements include sandbagging, closing doors, welding doors, hooking up pumps, and creating new circuits to reduce the risk from flooding. In two submittals, development of severe accident management guidance to reduce the risk of high winds is being considered. Hardware improvements include, for example, modifications that enhance flood protection.

- Potential failures of upstream dams, leading to flooding at the site, were considered and screened out in many of the first 24 submittals. However, generic dam failure data has been employed in all cases without considering site-specific information such as dam type and vintage.

- In general, the CDF contribution from HFO is lower than that from internal events.

Generic Issues

The IPEEE program has addressed a number of generic issues (GIs) and unresolved safety issues (USIs) including USI A-45 ("Decay Heat Removal Requirements"), GI-131 ("Potential Interaction Involving the In-Core Flux Mapping System at Westinghouse Plants"), GI-57 ("Effects of Fire Protection System Actuation on Safety Related Equipment"), Sandia fire risk

scoping study issues, and GI-103 ("Probable Maximum Precipitation [PMP]"). Some key observations from the review of the first 24 IPEEE submittals include the following:

- In general the seismic and fire evaluations of the IPEEE are capable of addressing USI A-45, without any special additional considerations. Also, no HFO evaluation reported any open issue pertaining to USI A-45.

- For most applicable plants, GI-131 had been addressed through earlier upgrades and analyses. Some IPEEEs evaluated the capability of the in-core flux mapping system for beyond design-basis seismic loads consistent with the IPEEE review level earthquake (RLE).

- Almost all licensees have followed the guidance in FIVE pertaining to the evaluation of the fire risk scoping study issues and GI-57. In a few cases, seismic-fire outliers have been noted. No submittals have reported risk-significant findings associated with either the fire risk scoping study issues or GI-57.

- Most submittals addressed the effects of increased PMP criteria with respect to roof ponding and flooding due to intense local precipitation. In all such cases. the impacts of GI-103 were found to be accommodated by the existing plant design.

- The IPEEE submittals also provide information relevant to some other generic safety issues (GSIs) even though the submittals were not explicitly requested to treat, and the IPEEE program was not originally intended to resolve, such issues. These issues include: GSI-147 ("Fire-Induced Alternate Shutdown/Control Panel Interaction"); GSI-148 (Smoke Control and Manual Fire-Fighting Effectiveness"); GSI-156 ("Systematic Evaluation Program [SEP]"); and GSI-172 ("Multiple System Responses Program [MSRP]"). The IPEEE review process has identified the extent to which the submittals provide information relevant to these GSIs, and how these issues can be considered to be resolved.

QUAD Cities Fire IPEEE

Although not part of the first 24 IPEEE submittals reviewed, the Quad Cities fire IPEEE submittal review has revealed some particularly significant perspectives related to fire risk. A brief summary of the licensee's fire IPEEE process and findings is provided below.

The licensee's fire assessment employed EPRI's Fire-Induced Vulnerability Evaluation (FIVE) method for initial screening and EPRI's Fire PRA Implementation Guide for detailed evaluations for screened-in fire areas. These evaluations include: assessment of individual sources that can damage safety targets (i.e., safety shutdown equipment); identification of fire scenarios, taking into account fire features, such as detection and suppression; determination of conditional core damage probability for the specific fire targets; and calculation of a scenario-specific core damage frequency (CDF) value. Additionally, multi-compartment fire scenarios were considered in the event that the fire barriers credited in the single compartment analyses are unable to prevent fire propagation in adjacent compartments. Walkdowns were also

conducted by Quad Cities plant engineers together with supporting contractors in order to: verify the compartment data; assess the seismic/fire interactions; identify the potential fire sources, safety targets, and locations of fire detection and suppression systems; and inspect the fire barriers.

The licensee estimated a total fire CDF at the Quad Cities to be about $5x10^{-3}$ per reactor year (RY). The licensee reported that the top five accident sequences, which contributed about 40% of the total fire CDF, were all in the turbine building involving postulated oil fires. The licensee stated that, even though the plant is in compliance with the NRC regulations, the lack of separation of certain cables in the turbine building and the complicated procedures needed for recovery actions were responsible for the high CDF number. The licensee used Nuclear Energy Institute's (NEI's) severe accident vulnerability criteria (e.g., CDF exceeds $1x10^{-4}$ per reactor year) and identified fire at the plant as a potential severe accident vulnerability. The licensee has implemented an interim alternate shutdown method involving the use of an independent back-up power supply for both units at Quad Cities to reduce the fire CDF from $5x10^{-3}$ per reactor year to $7x10^{-4}$ per reactor year. Currently, the licensee is evaluating long-term measures to further reduce the fire CDFs and is keeping the staff informed about its progress.