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SUBCOMMITTEE ON TMI-2 ACTION PLANS

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE ON TMI-2 ACTION PLANS

Room 1046
1717 H Street
Washington, D. C.

Thursday, October 29, 1981

The Subcommittee on TMI-2 Action Plans met at 8:35 a.m., William Mathis, Chairman of the Subcommittee, presiding.

ACRS Members Present:

William Mathis
J. Ebersole
D. Moeller
J. Ray

Designated Federal Employee Present:

R. Major

D. Fischer

Nuclear Regulatory Commission Staff Present:

G. Lainas
D. Verrelli
J. Beard
J. Lyons
D. Crutchfield
W. Shields
R. Cleveland

P R O C E E D I N G S

MR. MATHIS: The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on TMI-2 Action Plans. I am W. Mathis, Subcommittee Chairman. The other ACRS members present today, on my left, are J. Ebersole, D. Moeller, and J. Ray; and Mr. Ward will be joining us later on this morning.

The purpose of the meeting this morning will be to be briefed by the NRC staff on a proposed rule to 10 CFR 50, "Licensing Requirements for Pending Operating License Applications." This afternoon the Subcommittee will meet here again starting at 1:00 p.m. to discuss the systematic evaluation program.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act. Mr. R. Major is the designated federal employee for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the Federal Register on October 9, 1981. A transcript of the meeting is being kept and it is requested that each speaker first identify himself or herself and speak with sufficient clarity and volume so that he or she can be readily heard.

We have not received either written statements or requests for time to make oral statements from any member of the public.

We will now proceed with the meeting and I will call upon Gus Lainas of the NRC staff to begin.

MR. LAINAS: Good morning. I am with the Division of Licensing, and with me to share this presentation is Dave Verrelli on my right, who is also with us in the Division of Licensing, and J.T. Beard on my left, also with the Division of Licensing.

We are here to talk about the status of the proposed rule for the implementation of the TMI action plan items for operating licenses. As you know, these are the items from 0667, the TMI action plan, that were approved by the Commission for implementation.

If you don't mind, we won't use vugraphs, but talk from the handouts themselves. Generally, the outline of the presentation will be, we will give a background of how we got to where we are and we will talk about the content of the proposed rule itself and discuss some of the responses that have come back since the rule was noticed. In addition, we will conclude talking about the plans and schedules for where the rule is going to go.

MR. MATHIS: One other question, Gus. Are you going to touch on a similar rule for OL's?

MR. LAINAS: I'm going to limit it to OL's. This is for operating license applicants.

MR. MATHIS: I know, but the other is still behind us. It's put to one side, anyway.

MR. LAINAS: I think this might come out in the first slide.

MR. MOELLER: The material that was given us to be read says for the operating reactors it will be cut in a week or so. Has that week gone by?

MR. LAINAS: I think we're going to try to show you where we are, okay. The first item was March 10, 1981. During the review of the CP rule, it was decided between the staff and the Commission to develop a similar operating license rule to be applied to operating license plants. The staff went back and developed this proposed rule and completed its action on April 17, 1981.

On April 30, '81, the Commission approved a proposed rule for OL's, and also there was an agreement to go on with the rule for operating reactors. The rule was published and was noticed on May 13, 1981, with a 90-day comment period, that is the OL rule.

The proposed OR rule for operating reactors was developed by the staff in June of this year. The Commission in August -- it says Commission decision on proposed OR rule -- during discussions decided not to go forward with an operating reactor rule.

The comment period for the proposed operating license rule ended in August of 1981. Since that time we have gotten responses from the members of the public, licensees, applicants, and we are prepared to discuss what those responses look like, the comments.

MR. MATHIS: Good.

MR. LAINAS: So that sort of leads you into where we are today with the OR rule. As far as the content of the rule, I think just a very brief description of some of the things that are in there. The general objective of the rule is contained in there as described, and it states the Commission determined that the NUREG-0737 requirements should be codified into Commission regulations. It was felt that codifying regulations simplifies certain things. One thing is the enforcement. It makes enforcement a lot easier.

The other, it finally puts to bed any arguments or discussions, at hearings or otherwise, as to the technical validity of these requirements. In other words, you argue but once.

The intent of this rule was not to change the technical content of NUREG-0737, which was approved by the Commission. The rule itself was attached as paragraph F to Part 50.34, contents of application and technical information. Essentially, it was broken into two parts and it was those parts that would be immediately effective at the time the rule was issued. And there were also items in there which were dated items, we called dated items, which gave later implementation dates. And the intent of that was to really parallel NUREG-0737.

If you remember, that had two parts to it. The first part was the requirements and schedules for operating reactors. The second part was, what do you do with operating license requirements. So the rule paralleled that.

And the way it was broken down was, for those items in 0737 that were intended to be implemented during fuel load, those items were broken down to be immediately effective in the rule. So the other items, the dated requirements, were separated into a second part.

The way the rule is written now is, we estimate that the rule will become effective, I guess this summer some time. We predicted when it would come out, so we broke the two parts accordingly, paralleling the parts of 0737.

MR. VERRELLI: Basically, for operating license applicants we used a cutoff date of July 15. Anything that came after that was a dated requirement and anything before that was in the first part of the rule. You have to do this before you get a license. That was the cutoff because we felt it could not be effective before then.

MR. LAINAS: As we progress with the development of the rule, this may change. You will see that.

MR. MOELLER: I have two comments I think are appropriate at this point. One is, the proposed rule is not dated. And I have come back on the NRC -- looked at the NRC files on other proposed rules and none of them are dated. I would enter a plea at this point to please put a date on these things.

It says 10 CFR Part 50, licensing requirements for pending license applications, proposed rule. And there is no date on it. And if you look at 10 CFR 100 draft, there is no date on it.

MR. MATHIS: Dade, what you are saying is you need a date on the draft so you know which version you are looking at.

MR. MOELLER: Oh, sure. How do you keep up with them?

MR. BEARD: The date that we use is the publication date. In other words, it was published March 13th of '81 as a proposed rule. It's been approved by the Commission for publication in order to obtain public comments. We consider that as the proposed rule.

MR. MOELLER: Right, and if we have a copy of it as published in the Federal Register we also have a page number, volume, and a date. But you give us these typed drafts and you never date them, and it's very frustrating.

MR. BEARD: I'm sorry. I guess --

MR. MOELLER: Let me hold it up. Where is the date on this report?

This isn't for this Subcommittee meeting, but another one says, "This action will be implemented on July 4, 1977." So when I read that I assumed this must not be the most up to date draft.

MR. BEARD: Was it attached to a Commission paper?

MR. MOELLER: No, it's not -- well, yes. This is attached to a transmittal memo. But you don't always have the memo attached. And the memo now says something that I don't understand in terms of what you just told us. Let me quote: "This evolving process has led the staff to conclude that it should consider changing a number of requirements of NUREG-0737. It would be the staff's intent to thoroughly conduct a re-review of each item in NUREG-0737 during the public comment period on the proposed rule and make appropriate changes in the final rule based on such a review of the public comments received."

That helps a little bit, the second sentence. But you have just told us the rule is to implement or parallel what's in NUREG-0737. Well, if it's still under review and still being changed, then we have a moving target.

MR. BEARD: We are going to cover that subject in just a second on the agenda.

MR. MOELLER: Well, do we get someone to start dating these things?

MR. LAINAS: We'll make a note of that.

MR. BEARD: It is a good comment.

MR. LAINAS: Okay. Sort of parallel with that comment, in developing this proposed rule the Commission has recognized there are a number of items from 0737 that merit additional consideration prior to being implemented in the final rule. That's item C on the handout.

There were certain things that we recognized at the time of the issuance of the rule where changes might be made. They really fell into four categories:

The first was with respect to sufficient information received, and that is in the time it takes to issue the rule, the ongoing time, we are getting a better idea, we are getting information being submitted on operating reactors. So as a result of this, the rule may indeed change.

The second is that some items with respect to emergency preparedness and support facilities, there is ongoing regulatory action. Some proposed rules have been issued, and therefore the intent would be not to duplicate that in this particular rule, or at least recognize it.

The third is that some of the positions and recommendations in 0737 may be revised as we gain additional information. I have to say that each of these changes have to be approved by the Commission. Reconsideration and the need for modification; again, this is related to getting additional information and understanding the problems better. This will be reflected in the final rule.

The last item that's recognized is that some of the items are perhaps entirely too detailed, like specifying the failure modes and effects analysis for ICS. That may be getting into more detail than you should in a rule. And certainly with the manufacturer of the PORV, which was identified in the rule.

So these are the things that are ongoing today.

MR. RAY: Do you have any idea when these revisions might be effected?

MR. LAINAS: We have a schedule. That's the last slide. I don't know if it's worth getting into perhaps now.

MR. RAY: Take your time.

MR. LAINAS: Okay.

MR. BEARD: I wanted to add something along the lines of what Dr. Moeller commented on. That is, the information on the slide that was just presented is the information that was developed in the process of the staff developing the proposed rule that was put in the Federal Register. In other words, we were saying that we developed a rule and it's got a lot of items in it, some of which we want to reconsider before we go final with it ourselves.

MR. VERRELLI: In that sense, we solicited industry and public participation. So in the preamble, if you have the notice, it lists a bunch of items and examples of these five categories. We call out these and say, hey, public, give us your view on this approach.

MR. MATHIS: This is one of the things that has given us a problem in reviewing this, and that is there seem to be so many loose ends throughout all of the 0737 makeup. But you have to say, well, why a rule now? Are you ready for it or are you just trying to get ahead of the gun?

My own impression so far is that you are not ready. That's why we're here today. Maybe you can convince us.

MR. LAINAS: I think that's why we put some of these caveats in there when we issued the rule. I think you're right, there were some items in there we felt would probably change and we highlighted that, and that's what we did.

But you're absolutely right. I think some of those loose ends are indicated in the rule.

MR. MATHIS: Okay.

MR. LAINAS: I guess for your information what we did was, we tried to break out or at least list those items that were in the first part of the rule, that is those items that would take effect when the rule became effective.

In the second part of it, page 5, are those items which would be implemented -- which are what we call dated requirements. Now, again this is paralleling NUREG-0737. The note on the bottom of it, these dates might change depending on when the effective date of the rule is.

Now, Dave Verrelli is ready to go in and discuss some of the comments that we received, and to this end I think we have some additional handouts. I don't know whether you received them or not.

MR. VERRELLI: Jim's got them.

MR. BEARD: Do you want to give them out now?

MR. LAINAS: Sure.

What we did was take each part of the requirements and list what the public comments were, so you have that for your information.

MR. VERRELLI: As you can see, we are not at the decision point as to where we might go. This is really a status report of the types of responses we got from industry. I have a slide here that says "responses." Basically, we had 49 responses and I tried to give you a feel for the number of commenters we had. 37 from applicants were licensees; they have both an operating plant and they are an applicant.

Then I tried to break it into vendors, architect-engineers. And then there are others, such as the AIF, UCS, NRDC, and citizens, if you will. So it gives you a distribution of the types of responses.

Generally, most of the people were against rulemaking --

(Slide.)

-- for a number of reasons.

MR. MATHIS: That was no surprise.

(Laughter.)

MR. VERRELLI: 39 of the 48 or 49.

MR. LAINAS: Excuse me. Let me interrupt you. Maybe I should have made the point that the Commission in August of '81, when they decided not to proceed with an operating reactor rule, their fundamental problem with it was a lack of flexibility, and that's pretty consistent with the comments that we are getting now.

MR. VERRELLI: One of the main comments was lack of flexibility. This is both technical and schedule. Schedulewise, an applicant is ready for his license and the way the rule is written he would have to have certain things installed. And he may say, I can't get the equipment on that schedule, I don't really need it until I go to high power operation, so therefore I lose that schedule flexibility if the rule were effective in this form.

Technically, people like Westinghouse, an NSSS vendor, said, look, if you codify these requirements you remove some technical flexibility on alternative solutions. Maybe the solution that you have written in NUREG-0737 is not the best. So now if I have a different alternative I now am in a legal chain as well as a technical discussion between the staff and the engineers. And that's what we meant by the lack of flexibility.

Another argument is, it negates previous agreements. An example of this may be, take an operating applicant now who is in a hearing and maybe a Hearing Board has made a decision that may be less or more or alternative to what NUREG-0737 says. So if you codify just 737 you may be negating things that you had agreed to before based on staff evaluation, analysis and SER.

The third area of comments, duplicates other parts of the regulation. We have things like operator training and emergency planning, and the public said, why put it in two places? You stand the risk of ambiguity of differences. The operator who is applying for an operator's license now has to look in places other than Part 55. If you need this requirement, maybe you should have it in a different form. And that's what they mean by it duplicates.

Other examples are Appendix K versus the item in II.K -- really, that's a part of that common reference level for boilers. They say that's really a part of GDC-13. So it's that type of duplication, is the type of comments we have.

The third one was too detailed. Mr. Lainas gave you the example of this particular PORV that was tested and going to be put on McGuire, I believe, that failed. So anybody that wants to use that type of PORV has got to justify it.

Certain items they say have been resolved. A number of your items like II.K.2, which was the B&W orders or B&W provided the analysis, they say, you provided the analysis, the staff has issued an SER, the item is closed, why clutter up the regulations on something we should not have to provide additional information in the future? Such as, in 1990 why should an applicant for a B&W plant submit that information, that type of argument.

We also received the comment that the acceptance criteria is not really finalized by the NRC. And to give you an example, say the human factor aspects of control room habitability. This is already in the regulations, but maybe not all that NUREG-0737 is looking for.

Those are the general comments of the people who argued against going to rulemaking. I think it's only fair that there were four -- three utilities and one private citizen, or one citizen, if you will -- who were for rulemaking on 737. Their arguments are:

One, yes, it looks like the NRC is trying to enhance safety.

Number two, it defines the things we applicants may have to satisfy, as opposed to additional things coming down the road. In certain cases they say it would reduce the number of litigations. Let's litigate it once; let's not do it 13 different times. Those type of arguments.

There were five that we classified as not really for or against. In other words, we did not want to interpret whether they were for or against, but they commented on the rule and said it should be there or it shouldn't be there, but we didn't want to classify them as for or against rulemaking.

For example, the BWR Owners Group we classified as not for or against. They gave us many detailed comments and said, here are advantages or non-advantages. But they specifically said, we are not going to take a position, it should be licensees' position and we don't want to address it.

Now, I think it's important to cover the types of comments we got. I have listed ten of them here. Basically, you have a list of the comments, but I thought t ey would be worth covering.

The first two, we got a very large number of comments on staffing. We received 60 comments. On staffing there are two problems licensees have: One is on the manning; the other is on overtime.

In manning, they say the shift technical advisor was intended to be an interim solution pending the upgrading of operators; therefore, it shouldn't be in the rule. They are saying the STA was required to be on call and be there within ten minutes. He was required to be there for consultation. Why should I put him in a manning table as being on shift? That's a change in requirement.

On the emergency response, where there is tabulation after an emergency, different types of people have to arrive at different times. They say in a remote site that's kind of difficult, for people to get there within ten minutes or 30 minutes; it really ought to be an hour.

They also made comments that the certification of non-licensed people should be eliminated from the rules. In other words, the certification of health physicists and things of that sort.

The second most commented on was training, number two. I guess basically most of the people said, we agree on upgraded training, but don't put it in the application, put it under Part 55. So that again, an operator who's applying for an application, whether it be an RO or SRO, knows just exactly what's to be satisfied in the initial license in the sense of retraining.

MR. MOELLER: Excuse me. You mentioned overtime in the staffing. Could you refresh me on the controversy there?

MR. VERRELLI: On overtime limits NUREG-0737, which is basically a copy of the letter we sent to licensees back in July, as well as the rule, says you should limit overtime. And the guidance says basically you shouldn't work more than 12 hours.

You're going to get a lot of good comments on this from actual people who have experience. So that's why I referred to licensees, even though we are talking about OL's. They say, I don't work people a shift and a half, I limit overtime and I make special provisions as to when I will authorize overtime. But I can't really live with 12 hours; I need 16. I'll take compensatory measures some other way. I'll make sure he's off three days or four days. Alternatives to what the staff criteria was. The staff has not really finalized the criteria. We are accepting these comments to see what we should do in the overtime area.

MR. BEARD: Let me add a comment if I may. The staff fully intends to limit overtime, there's no doubt about that. It's obviously in the best interest of safety in plant operations. The issue today is should it be 12 hours, 14 or 16 hours. After a man has worked overtime, let's say for two days, should he be allowed off or required that he be given off 24 hours, 28 hours, 30 hours for crew rest, that kind of consideration.

So the only issue at this point is what should be the best numbers to achieve a good improvement in safety, but at the same time not interfere with the man's capability to operate the plant.

MR. RAY: Is the 14 and 16 in our example that you mentioned a containment of superior service or is that accumulated over each period?

MR. BEARD: No, we're talking about a case where a man has worked a shift and he's required to stay over, a double shift, if you will. If I have to double shift him, I would like to, since the union in most cases requires me to pay him for a full shift also, I'd like him to work a full shift.

MR. EBERSOLE: What's the airline companies' requirements on this for pilots and so forth?

MR. BEARD: I can't give you a specific answer other than just to tell you off the top of my head, they have limits that relate to the duration of the flights. For example, if he's going to fly from Boston to New York to D.C. to Miami, that may be a 12-hour trip.

The key thing is to have general limits on that, on the flexibility, but more important in my experience is, give them plenty of crew rest before they go again. And we are considering the approaches used by other agencies here in town.

MR. EBERSOLE: Well, however you get that accomplished to keep on being effective, it's arbitrary, isn't it?

MR. BEARD: I personally think so, sir. You can make a case that says 12 hours is safe and 13 hours isn't, but that's hard to defend.

MR. VERRELLI: You have to be concerned what the operator is doing. You would never let him sit 16 hours at the control board. You have to have some relief and rotate those people. But these are alternatives, again.

MR. MATHIS: Are you closer to some finite numbers now?

MR. VERRELLI: I don't think we're prepared to answer that now.

MR. BEARD: The answer is we are definitely closer.

(Laughter.)

MR. VERRELLI: What did I say, we were making progress?

Did I answer your question on manning?

MR. MOELLER: Yes.

MR. VERRELLI: The next item was the small break LOCA problems, and you're discussing specific tests, and they really should be in Appendix K. There is really no difference. Just a tuning of Appendix K.

The fourth item there is for cooling. I'm sure everyone knows that's the reference to water level. The licensees are saying two things, two types of comments under inadequate core cooling. Applicants say, number one, water level may not be the best answer, it may be ambiguous, and it really may add nothing.

The second thing they said is under this item you have a lot of training to recognize inadequate core cooling, and again, this belongs someplace else, not under the contents of the application.

Independent safety evaluation group, the next item. This is very similar to comments they made on the STA. They say this was really to be an interim requirement, it was to be tested for a year or two years and then re-evaluted, reassessed, and should not be in the regulations at this time. We are not prepared to defend that.

Containment isolation, not surprisingly is on the list. The way the 737 is written and is published, where it says that all non-essential systems should be isolated, applicants and licensees argue that there are cases where certain systems should not be isolated. The second part deals with the radiation signal to isolate purge valves. It was principally comments from BWR's, not all BWR's. But the majority of them say we really satisfied the intent of containment isolation without a radiation signal.

MR. MOELLER: Excuse me, is that where you isolate on safety injection?

MR. VERRELLI: No. In the BWR case, the way the requirement reads is that for containment isolation you must have diverse signals. One of them must be radiation, radiation before it is released. So if you have radiation in the dry well, you will isolate those purge and vent valves on that specific signal.

In addition, you have others. You have the water level containment and the pressure.

Position indication received a number of comments. I classified them basically as editorial. People were saying the words are in 737 it says direct indication of valve position, and really you mean a positive indication because you have accepted things like acoustic monitors. That's not a direct indication.

MR. EBERSOLE: That was limited to PORVs. Was there any indication about any valves that needed better indicators, other than PORVs?

MR. VERRELLI: No. We have valve position indicators or relief valves and safety valves also.

MR. EBERSOLE: I'm talking about other kinds of valves, standard MOVs.

MR. VERRELLI: Standard MOVs.

MR. EBERSOLE: You had no comments?

MR. VERRELLI: No. Lots of disk failures and shaft failures and so on.

I don't recall that.

MR. BEARD: There were no comments addressed to this rule that brought up that. I think it was a good point but it was not addressed.

MR. EBERSOLE: I think there is a remark I'd like to make. Lots of things are sharp-pointed toward one objective but they have generic significance, and this is a case in point.

MR. BEARD: I have to agree with you absolutely, but let me remind you of the problem we encountered at TMI. The problem was that they had position indication. The method where they chose instrumentation was, as you probably remember, was the tailpipe temperature monitors.

What happens is, once you blow the thing, the temperature goes up and stays up because the tailpipe stays up. But what we're trying to arrive at is a better method that will give you more positive information. Is the thing really closed, and is the temperature really hanging up or what, to give the operator less confusion to deal with.

MR. EBERSOLE: Can't you do that by core design?

MR. BEARD: I would choose not to give a response to that.

MR. EBERSOLE: Okay.

MR. RAY: I'm not clear on this requirement. I'm looking at item 12 on the details, and it reads, "Direct position indications open or closed for the relief and safety valves shall be provided in the control room." That doesn't say it's only in the pressurizer. That's your intent? It's all valves?

MR. BEARD: Yes, sir, it's not all valves in the sense of like your MOV's that are in, say, the emergency core cooling systems which only transport water. We're not talking about those valves. We are talking about PORV's and safety valves.

MR. EBERSOLE: And to that extent, it is ambiguous.

MR. RAY: Then this comment that it's not clear is correct.

MR. VERRELLI: This is the Lessons Learned Category A requirements that people had to put in.

MR. EBERSOLE: TMI generated a lot of reactions where people do things along narrow lines, so I'll look on this kind of like you used a loaded canon, it's full of huge pieces and little bitty pieces and so on, and a lot of it has a stamp of being a compulsive generation of things that ought not to have been done. It's a bunch of patches on problems.

MR. RAY: Corrections after the problem.

MR. EBERSOLE: Right. And I hate to see a rule that perpetuates patches, rather than a rule that really gets to the bottom of the fundamentals of why these things really occurred.

MR. BEARD: I think it's a valid comment. I think the Commission has taken a position along with the staff that as a package underlining the package, that this is a substantial improvement to the public health and safety. Individual items could be yea or nay.

MR. EBERSOLE: But to an extent they're compulsive, running up and patching holes in the barn; they don't really get to the roots. You read them over one time and look at the variation and substance and content and scope of each one of them. There's an infinite variety of these things.

MR. BEARD: There is some order to the madness that may appear. The requirements in TMI documents are arranged and categorized into separate categories. The first category I might remind you of is called Operational Safety and deals with administrative matters, training and management.

I might also remind the subcommittee that this document, 737, originated from a steering group which created 0660, the task action plan, or the TMI Action Plan. Now, that came from an enormous set of recommendations and studies by very diverse, separate groups ranging all the way from Congress, special inquiry groups, every citizen who wanted to cast a vote, just about. And it was a very huge assortment.

One of the problems that the steering group faced was how do you take these requirements, a number of which you may essentially get the same requirement from a number of different reports, but they are said a little bit differently. The slant seems to be different, and how do you get a handle on this. Furthermore, how do you decide which requirements are worth implementing and which are not?

You know, one of the big decisions was do away with it, the Commissioners, and get a single administrator. So I guess the point I'm trying to get across is your objective is a very good one. I wish we were there, personally. We've come a long way.

MR. EBERSOLE: Well, I read these things -- you know, I have to read these. I say what is the extrapolative content of this, what does it mean? Or broadly, what should I do beyond the scope of what I see here on each of these points that have been made.

MR. LAINAS: You know, these are very discrete items and there's a weave going through them that says do a modification to upgrade your aux feedwater that really follows in schedule after an evaluation. It's implied that I'm going to get an evaluation and will review it and not just patch but solve that problem.

MR. EBERSOLE: Lots of these I saw say do it on B&W or Westinghouse or boilers. Really, the substance of the comment, though, is it can encompass all reactors.

MR. BEARD: One of the reasons why you find that is you may find a requirement that says do it on boilers, because we do it on the B&W's immediately on the Westinghouse and on the PWR's promptly. And now 0737 in a more deliberate way says well, some want to look at the boilers.

So the applicability is limited to maybe boilers because everybody else in the nation has already done it.

MR. EBERSOLE: Don't you read this as though this is a thing that stands as a -- what I'm saying is it's not just a point in time. This is going to apply over the long term.

MR. VERRELLI: Over the next 20 or 30 years.

MR. EBERSOLE: So I say what happened to the boilers or Westinghouse? I read something in here about cooling by injecting crude oil and design engineers don't have design tolerances for that.

MR. VERRELLI: But they may have in the future.

MR. EBERSOLE: There are plans out now. I recall we just issued -- I can't think of which one; it doesn't intent to even conceive the idea that feedwater is ever going to be, period. It's not making any provisions for primary coolant flow as the primary coolant. Which one is that? Is it Offshore Power Systems? I can't remember.

MR. MATHIS: Jesse, I think the problem you're pointing out here is it's just a bigger problem than what's covered here. But it's aimed directly at 0737 and the TMI Action Plan and it's not all-inclusive. And I guess for the time being we have to accept it that way and say okay, this is a step.

MR. EBERSOLE: I guess we'll go over point by point anyway.

MR. LAINAS: I don't want to leave this because I don't think you're trying to make a point -- I don't think we're trying to make a point -- that these are extraneous requirements. I think generally these were lessons learned from the TMI accidents, and I think there are two problems.

First of all, the detail that's required; and secondly is the implementation dates. We have been talking to utilities, several utilities, in trying to understand what their problems are, not only with respect to the implementation of these particular TMI items, but I've heard the expression "global considerations" of not only these but the requirements from bulletins and orders, I&E requirements. And I guess you've seen the version of those by the recent reorganization in the Executive Director's office.

So I think we're trying to be reasonable and rational by applying these things, but I think it should be made clear that I don't think anybody in this room thinks these are extraneous. I think they're well-founded. I think it's just a matter of the implementation time.

MR. MATHIS: First, there's the one problem that I think we all recognize as difficult, and that's the one as far as inadequate core cooling is concerned, and how to measure it. I'm sure you're aware that we have written a letter on that subject questioning how you do it. There has never been a clearcut answer from the utilities or from the vendors as to how it can be done.

Now, how are you going to cover that particular topic in this rulemaking? Maybe this is not the time to ask for this.

MR. VERRELLI: You're asking for a resolution? We're not there yet.

MR. EBERSOLE: Are we going to take up that item on its own later as a topical discussion? Is that the plan of action about the level problem or whatever? Are we going to go down the line here and take these things up?

MR. LAINAS: We're not really prepared to go into the technical aspects of this requirement.

MR. EBERSOLE: I have some problem with the fact that people can't believe the level gauges.

MR. VERRELLI: But yet, we received that comment. That was one of the comments we received.

MR. EBERSOLE: I heard the comment 50 years ago, and it was offered as an escape to not put these things in, and I see it still is.

MR. MATHIS: Apparently we aren't going to get into that kind of detail here today, so why don't you go ahead, Mr. Verrelli.

MR. VERRELLI: The next three items I just want to mention to give you a feel for the number of comments on the specific item, and those were the reference water level that I discussed earlier. And people likened the boilers, and as I said, it's already GDC-13.

On emergency procedures again they're saying it's part of the rule. You have Appendix C; why don't you modify that and don't put it here.

And the final one is really an administrative one on the problem of reporting leak valve failures and challenges. They say that's what the LER system is all about, so why here? Those are the types of examples on those types of items.

A couple of general comments that we received such as from NRDC and UCS I thought you might be interested in. I categorized them as against rulemaking for the following reasons. The arguments were, number one, this rule is not adequate and not sufficient for a resumption of licensing that new plants can be operated in a safe and efficient manner. And they discussed a lot of the development items under NUREG-0660.

Secondly, they said it appears that the Commission is attempting to prevent litigation in individual cases, and we don't think that's right. Thirdly, I characterized them as saying there is no basis in this regulation for this particular requirement as to why it is required and why it is okay for the acceptance criteria.

That is the flavor of their response and I thought it might be important to mention that.

From Westinghouse as an NSSS vendor they said you shouldn't go to rulemaking. How about those plants that are currently under review? What you're going to do is delay the licensing process and negate previous agreements, along with other such arguments as duplicating roles.

I mention also that the BWR owners group had detailed comments on each item but refused to take a position.

A number of examples on the boilers where they said we tried to solve this problem generically through an owners group as opposed to going to individual applicants or licensees. We think we are making process. The staff has not yet accepted overnight solutions, and therefore, we think going to rulemaking is going to complicate that issue.

I have attempted this morning to try to give you the types of responses we got in from home. The commented on issues, and from that I think we have to say well, where are we and where do we go from here.

MR. BEARD: Before we get into Mr. Lainas' talk, I think we may have given you a misimpression of what the situation is. By that I mean we have told you that the vast majority of the commenters say we are against rulemaking. There still seems to remain a need to establish an enforceable regulatory basis for requirements that previously did not exist. I think in my own personal view, therefore, some rule will be arrived at and many requirements that will contain what detailed specificity, and what else will be in it I can't say at this point. That requires Commission action.

But I don't want yo to go away thinking that our presentation here is that we are suggesting in some way that the staff or the Commission not go forward with rulemaking.

We are not making that suggestion today.

MR. RAY: But you have not indicated an unbending attitude. I gather from your presentation that you were open to some of the suggestions and may very well modify this rule significantly.

MR. BEARD: I think the answer to that, sir, is we want the requirements to be reasonable and enforceable. We want to avoid requirements where there are not any needed.

MR. LAINAS: This was specifically mentioned in the rule.

MR. RAY: The rule is not cast in stone, by any means.

MR. LAINAS: That's right. I don't know whether this has been done before, but this reconsideration is explicitly mentioned in the rule.

Okay, where do we go from here? A change to the rule -- that has been done, of course, as the rule did reference some forthcoming action on operating reactors. The Commission has decided not to go with the rule with respect to operating reactors.

MR. EBERSOLE: Could you elaborate on why you think that's the case?

MR. LAINAS: I think because of flexibility. I think that was their comment, that it took away some flexibility.

MR. RAY: Once again, things are never cast in stone and should never be cast in stone. Do you think this decision is just as of this time, or from the viewpoint of the need?

MR. BEARD: It's hard to say.

MR. LAINAS: I wouldn't comment on that right now. The thing is with respect to this OL rule, we're going to notice it. It has been noticed. But we're going to notice it again taking out the reference to the forthcoming operator reactor rule. And it's not out, but it's imminent. The noticing is going to have a 30-day comment period in it, and as indicated in the last slide, this is what our schedule looks like for the end of the comment period and the time it takes to resolve the comments and propose a finale rule.

(Slide.)

You will notice that the ACRS is included in it, and I think that a recent direction is that proposed rules will be discussed with the ACRS, and this will certainly be taken advantage of in development of this rule. And we plan to go to the Commission and see what their decision is and our current estimates are for March. And that's about it as far as this presentation is concerned.

What we thought we'd try to do is come down and explain the rule as best we can; maybe give you some of the background and tell you where we stand and where we're going.

MR. MATHIS: Gus, I have one question. Looking at the schedule after what we've heard and discussed here this morning, you've got a lot of loose ends. Are you going to get all of that, do you think, tied down by February? Or late December, really? That's kind of optimistic in my opinion, isn't it? Maybe I'm wrong.

MR. LAINAS: No, I think it is an optimistic schedule, yes. But I don't see us slipping by more than a month. I don't think it's that optimistic.

MR. BEARD: And you've got Christmas holiday, and that could be a deterrent to schedule.

MR. MOELLER: I had a number of questions that I would like to ask about. This is, of course, a legal document. I mean, the rule will be.

MR. BEARD: It is definitely a legal document. That's the reason why we're going into this exercise, is to put it into 10 CFR Part 50. It will become a federal regulation.

MR. MOELLER: Okay. With that in mind, I'm not a lawyer and would never claim to understand it at all, but are PWR's and BWR's legally refined? You know, you referred to them in here. Is there a legal definition, so I know which type my reactor is?

MR. BEARD: May I defer that to our counsel sitting behind us here?

MR. SHIELDS: I don't know, I don't think it's defined anywhere in the regulations as far as I know.

MR. MOELLER: And yet you apply certain things only to PWR's and only PWR's. Let me give you some more. You talk about B&W designed plants. Now, I could see a day when, you know, someone else might build a plant or take up some features of B&W plants in Westinghouse. Are B&W plants, designed plants, legally defined?

MR. SHIELDS: I don't think any of these things are defined in the regulations now. If there is any possible source of ambiguity as to this rule as we're going to final rule, it's possible that we can clarify it here. We've gone through this exercise recently in the CP rule which is not yet out in final form. And in that case, since the class was very limited, we actually listed the plants that were covered. In this rule, we could probably do the same thing by a footnote, I suppose, in the front, since it would cover persons whose application for an operating license had been docketed as of a certain date.

You can always put a footnote up front and say who is covered by it. I'm just not aware as of now that there is any source of ambiguity in the way those categories -- that kind of designation was taken from, as I recall, the NUREG, and I don't recall anyone noting that there was any possible ambiguity as to how the NUREG applied. But if there were such a thing, I think we could easily clarify it.

MR. BEARD: I think this kind of comment that you bring up is the kind of thing we will clean up in finalizing the rule. We feel like at the time it is a proposed rule and a non-legal document, and for the purpose of soliciting comment it wasn't necessary to polish the language to that extent. But we do intend to do so as far as I know before we finalize it. So there will not be these kinds of ambiguities.

MR. MOELLER: Like on page 27 you say, quote, "This evaluation shall consider the LOFT test."

Let me comment too on a few other things. I have certain, I suppose, idiosyncrasies in reading these things. I try to be as critical as I can be.

Like on page 7, you have to begin at the bottom of page 6.

MR. BEARD: Since we're all dealing with different versions, could you give us an item number? We're dealing with the copy that appeared in the Federal Register and looks something like this one. I'm holding it in front of you. I think what you have is a copy that was attached to the Commission paper.

MR. MOELLER: I have the copy of the memo that was attached to the Commissioners from Dircks.

MR. VERRELLI: What was published is slightly different.

MR. MOELLER: Okay. Were we provided what was published?

MR. BEARD: I happen to have ten copies here.

MR. MOELLER: Perhaps you have corrected this, but in this particular thing it says: "For example, there are several items for which the ongoing Commission review, based on submittals by operating reactors, may do," so forth. I'm nitpicking at words, but I don't know a single operating reactor that can submit anything. I know an operating reactor licensee that might submit something.

MR. BEARD: You're absolutely correct, sir.

MR. MOELLER: I'm trying to be helpful here.

Well, the pages mean nothing. Well, it's difficult to cite it, but there was a typographical error on page 17, line 8, of the draft that I am using. Let me see if it's on these others.

Oh, this one. Again I'm nitpicking, but it's talking about the control room and it's on page 29 of the draft. It says: "Analysis based upon the final as-built condition shall be provided to demonstrate that airborne concentrations of such hazardous fumes will permit control room operators to do" thus and so.

Therefore I am, I realize, extending my criticism to the ultimate, but I found I didn't know what a control room operator was. I know what a reactor operator or a plant operator is. You realize I'm nitpicking, but again I was looking at it as a legal document.

MR. EBERSOLE: Before you leave that topic, GDC-19 in its own right has been a monumental problem for the last 20 years. As long as we're going to bring it up, I think we ought to deal with it once and for all in many respects. I can go back 20 years and remember the arguments about operating the reactor from a point external thereto, and I imagine you could go out on the street and ask the man on the street, what does that mean to you, does that have any connotation that something's happened in the control room that should enable you to operate the plant from a distant point competently?

And believe me, the man will say, yes, it's burned out or blown up or destroyed, or fixed. And then you tell him no, it means. . . you get a derisive laugh from the general public, which you deserve. And yet that still stands.

And this, I get back into my general point. To fiddle around with a tiny point like GDC-19 without going in and fixing the damn thing I think is stupid.

MR. BEARD: I think that the question raised has been around for a long time. I am also sympathetic. I understand and am aware of some of the extremely astronomical costs of making it maybe the way you would like to make it because of those considerations.

But the thrust of my comment is merely, the operation that we are reporting on today was restricted very severely to taking a NUREG document with the requirements contained therein, translating that document into a rule. Now, we did receive one comment, for example, that says you ought to be including in the rule additional items.

I'm very sympath tic to that, and as we can get to those kinds of considerations I'm sure the staff will undertake those. I think your comment is very important.

It's a similar item. But I'm under restriction from the Commissioners that we do not add things to 737 that they have not specifically approved for implementation.

MR. EBERSOLE: Therein lies the basis for patching and you're patching a little part of 19. That's a problem and I think it ought to be called out and recognized, and we leave behind us chaos. We are like the national debt. We are wallowing in these types of things and we will wallow worse and worse as long as we perpetuate this process.

MR. MATHIS: We can't do very much about that either, Jesse.

MR. BEARD: I'm very sympathetic.

MR. EBERSOLE: I can go back to the Browns Ferry fire and why -- why many things weren't said about the fire which are very pertinent to GDC-19, and one of those is the extraordinary astronomical cost it was presumed to take to fix such matters. It's not astronomical. It's modest compared to the benefits to be derived from doing those things in this area.

The fact it was astronomical, it was typical of the fact that applicants tend to make these estimates in order to avoid doing things they ought to do, and they can convince the regulatory people that those estimates are real when in fact it's not.

MR. MATHIS: Jesse, that leads to one other question I had, and that is the deletion of the OR rule leaves something kind of dangling. Is there any more thinking as to, okay, here does that go from here and how does it intertwi ..th the OL?

MR ..RRELLI: Basically, your comment is saying if we're .. going to rulemaking for operating reactors, what method are we going to use to implement NUREG-737 requirements on operating reactors, is fundamentally what you're saying.

Most of the schedule in 737 for operating reactors will have come and gone by the time we even project this on an optimistic schedule, with the exception of I think there is a 1-1-83 date on emergency planning. There's only about one or two items for next year. So therefore you must have already taken some position, rather than going to rulemaking for operating reactors. And we are tackling that problem right now.

In looking at operating reactors, we have kind of broken them into various categories of things that are scheduled. And we asked licensees to do certain things, and things that we asked them to do come 1 January '81. We covered those and staff issued orders for them to do those items, and we are now looking at the rest of the items. And the staff will be recommending to the Commission maybe an alternative as to how we should be implementing these things in operating reactors, giving consideration, as Mr. Lainas said, to all the other requirements that they had.

MR. MATHIS: But those same comments apply to OL's pending.

MR. VERRELLI: But we should have solved the problem for OR's long before that. I just offer that for a rationale, as opposed to waiting for a rulemaking on operating reactors.

MR. MATHIS: I guess my question is, if you can cover OR's as they are today and the pending OL's are covered by the same action plan, then why isn't that sufficient for tomorrow?

MR. BEARD: You need to establish the basis for the regulatory applicant who walks in the door in 1992. He needs to be informed that there was an accident and there are no requirements, and from a legal viewpoint we need a vehicle to tell him, this is required when you apply for a plant CP or OL in 1992. We can't depend on a NUREG item way back there.

MR. MATHIS: I hear you, but --

MR. EBERSOLE: The applicant will always say, you should give me more room in which to make intelligent decisions. Don't confine me to the narrow confines of the rule. And he will then tell you later on that you weren't specific enough to keep him out of trouble.

So you bounce between these two walls. So within the experience you get for operating reactors you must for a better basis for operator rules for the coming plants like this. Would it not be prudent to wait for your experience and warn the forthcoming applicants to read the papers and see what's going on and come forth with their conservative interpretations and see if they really mean what they say?

MR. VERRELLI: This is one of the reasons for recommending an additional 30-day comment period, because there may have been licensees and operating reactors waiting for that to offer us their substantive comments.

MR. EBERSOLE: In part, then, you are waiting to see what happens.

MR. VERRELLI: That's true. And the staff itself is doing what we said we were going to do in the sense of re-reviewing the schedule to see whether that item has been resolved or not, this is going on concurrently.

MR. BEARD: Another advantage I believe we have is the same people who are dealing with the operating plants for the technical alternative solutions and the schedule relaxations to get to the next refueling and the rulemaking process are all in one organization headed by Mr. Lainas. So we get the advantage of cross-fertilization in that way.

MR. SHIELDS: Let me mention one other distinction in terms of letting the operating reactor rule go by the board. The other two rules we were working on, which is this rule and the CP rule which the Commission has approved and not yet published, is that these two rules have direct effect on the hearing process in particular.

In the operating reactor context, you are not worried about what might happen in hearing. So at least I think part of the intent of this rule and the CP rule is to state what issues are available for litigation related to TMI and licensing hearings, operating and construction. In the operating reactor context, you're dealing with individual licensees trying to resolve outstanding issues, and a rule in some ways gives you some additional enforceability.

But on the other hand, you are not worried about a hearing dragging on for years over what should be a necessary and sufficient set of TMI-related items.

MR. EBERSOLE: We seem to swing between two bounds here. If we issue a rule we've got something definitive that you can avoid a lot of litigation. If you leave it in the other context we have here, then it appears it shows up and is a delaying and costly thing on individual licensing applications.

In short, any applicant will have to fight this thing out in the field, surrounded by probably superstition and emotion rather than factual content of the arguments more often than not. And it's not a very good battleground for reaching any very good decisions out in the boondocks where the hearing is being conducted.

Doesn't this suggest some other process ought to be developed where that sort of litigation could be made in an environment where the issue is taken up as an issue on a generic basis and fought out without burdening the individual applicant for fighting it out just for his cause?

MR. SHIELDS: I don't know of any other way. We only have a few ways of enforcing requirements in a legal sense. One is through the adjudicative process and one is through the rulemaking process. Outside of those two specific authorities the agency is given, we don't have any enforceable powers. We have to draw from one of those two sources, and in fact even the adjudicatory power has to be based on a set of rules.

The point of generic rulemaking is to allow the fight to be done just once for everyone and not to fight out issues in an individual case. What we found in the CP area was that as we went into proposed rulemaking, a near-final rule when everybody knew what was going to be in it, the people that were affected by it were ready to make their submittal satisfying the provisions of the rule long before the rule was even approved by the Commission.

And I expect that the same thing is true of people who are now in the operating license proceedings, that those applicants are aware of the content, the basic content of this rule and NUREG-0737, and knowing that nothing is going to be added, although some things may be subtracted, and are probably ready to make those submissions to boards where they are before a board.

So I suspect the delay that would be involved is really small, and it's possible that this could introduce, I suppose, some delay into the review process. But if these are going to be requirements during the review process, I'm not sure it makes much difference whether it's a rule or a staff position.

MR. EBERSOLE: Thank you.

MR. MOELLER: I had one other comment on a word, and I'm really not trying to be critical as much as to point out to you words that gave me a personal problem. In paragraph 14, very near the end of the proposed rule, you talk about a new design for the automatic depressurization system.

It's the next to the last paragraph, at least in the draft that I had, and it goes on after a couple of sentences. In paragraph 14 it says: "For operating licenses issued prior to April 1, 1983, the design shall be installed not later than" such and such a date.

Now, probably in the field the word "design" has a meaning. To me again, I would have said "the newly designed system must be installed" by such and such a time.

Now, in terms of broader, and I hope perhaps more constructive, types of comments, I had a couple. On page 7 -- again, it doesn't help you, but one of the things you called for is for the ACRS as well as all reviewers of the rule to specifically address the question: Are there items in the rule that should not be covered?

Did you summarize what your comments to date have said? What were the main items in the rule that the public or the licensees or so forth say should not be included?

MR. BEARD: Maybe I can address that. We had broken out, as you can imagine -- the stack of comments that we've got sitting over here on the left is a stack of material approaching six inches thick. The comments are from wide sources with diverse opinions on about 70 of the requirements, plus general subjects.

We did do another crosscut that identified the nature of the comments, such as -- I'm trying to give you a good example to more directly answer your comment. But let me give you one on the top of the page. The comment is the requirement is too detailed. Okay. The way we did our search was, how many times did we get that particular comment, and we got it 107 times.

We asked ourselves the following question: How many items in the rule drew that kind of comment? It turns out there were 25 items that got 107 comments that said either too detailed or inflexible, too rigid. We went down through them this way also.

MR. MOELLER: That really doesn't say, though, that we do not believe this item should be included in the rule, does it?

MR. BEARD: We did no crosscut, I don't believe, in the very broadest sense of where a commenter suggested the item be deleted. I can tell you from my own personal knowledge of looking over these comments that a number of them say, delete it from here because it's already covered, or addressed at least, in some other part of the regulations.

MR. MOELLER: So in those cases they are saying, don't cover it here, like you were saying for emergency planning, operator training.

MR. BEARD: Control room designs and the habitability or human factors. And I think again, you have to recognize that people giving you the comments have a certain bias.

MR. MOELLER: I find your comment interesting, inasmuch as it says: "Comment is specifically solicited on items that may not need to be included." You would have thought more of the people would have said, here is the list of items that we do not believe should be included.

MR. VERRELLI: NUREG-660 was drafted and then 737, which was shortly over a year ago. And at that time the document was submitted. There was about a year between that time and when we drafted this proposed rule.

When we drafted that we said to the staff, are there certain items that maybe should not be in there. And we received these examples that you will have in your paper, that say for various reasons maybe they should not be in the rule, based on our best judgment at the time. We have a report from an owners group or a submittal or the item appears to be resolved, and the staff had not signed off on it at that time.

So we tried to highlight those items, to solicit the public's comment on the staff's thinking at that time, recognizing they had not been resolved at that point. So that was the source of the list of examples.

MR. BEARD: What we found was they were scattered among the specific comments. They said, if they wanted us to delete these 16 items with no basis it wouldn't go very far. So what they did was say, for item number 12, was to say, delete this, it's already covered, and give a reference.

MR. MOELLER: Okay. That is helpful.

On page 16 of the draft that I have, you have the statement: "The training program for all operating personnel shall include training to recognize. control and mitigate the consequences of accidents in which the core is severely damaged."

I wonder, who are all operating personnel? In other words, should the health physicist know something about this?

MR. BEARD: Let me again try to address that one, and Dave can help me as I need it here. The intent was the operating staff as licensed. This comment drew a lot of flak that said, make it more clear you're talking about licensed people.

We're not talking about auxiliary operators running around the turbine building. We're not talking about HP's. We're talking about licensed people up to -- I guess I have to put one caveat on that, and that is it probably would include the STA, the shift technical advisor. But with that single exception, basically the licensed people.

MR. EBERSOLE: Dade, are you going to go to another point?

MR. MOELLER: Yes.

MR. EBERSOLE: Before you leave that one, I have some notations on that. Consistent with our concept that we ought to prevent and mitigate, I think that particular sentence reflects the compulsive nature of many of these things where we are trying to do things to, to accept TMI-2. So the emphasis is on recognize, control and mitigate in accidents in which the core is severely damaged.

Why shouldn't it say the training for operating personnel shall include training to identify the significant initiators of potentially serious failure, test gauges; and then, two, if they failed in that, do what you say, recognize, control and mitigate the consequences of their failure?

MR. BEARD: With all due respect, sir, having had some direct experience in training operators, there are people who would put forth the counterargument that they go through a very extensive and expensive two years of training, all of which is intended to do just that.

MR. EBERSOLE: I know, but that's our problem. Has it done that?

MR. BEARD: I think our track record for safely operating reactors is admirable.

MR. EBERSOLE: You mean you would still say that?

MR. BEARD: Even with TMI, it's my personal opinion, yes, that it's outstanding.

MR. EBERSOLE: I on the other hand would disagree that the ability to recognize cascade initiators is not well done, because operators are not given the engineering knowledge on how our systems are designed and the peculiarities of the systems as they affect each other. They are more often than not told which knobs to turn and which buttons to push.

MR. BEARD: You raise a very good point, sir. I'll be very glad to talk with you during the break and give you some of my personal opinions.

MR. MATHIS: Dade?

MR. MOELLER: On the bottom of page 18, again of the draft, you have a paragraph about, "A management system shall be provided to independently verify the proper performance of operational and maintenance activities as a means of reducing errors that could result in or contribute to accidents."

This is paragraph 8. Here's the sentence I didn't understand: "The system shall include automatic status monitoring or verification by two qualified individuals." Could you help me a little bit about that?

MR. VERRELLI: The question regarding the two qualified individuals, this requirement was the one where we said when you put a safety system back on the line you ought to have a double verification that it is lined up properly. And in the NUREG we said we really haven't made up our mind yet on what a "qualified" individual is.

So the obvious question is, do these two people have to be licensed or is one of them licensed and one of them just adequately trained on that system? That is not yet resolved and that is why you will see in the preamble that particular item, which is 1(c)(6), is particularly called out, because we are not really sure and we have not developed a position on what a qualified individual is.

Obviously we would accept two licensed people.

MR. MOELLER: Thank you. That's helpful. Just a couple more, Mr. Chairman, and then I'll be through.

On page 33 you cite the TID-1340 source term. This is under paragraph (a). It's paragraph iii, with the subparagraph (a), and you know fifty percent of the noble gases and one hundred percent of the noble gases and fifty percent of the halogens and so forth.

Now 10 CFR 100 is also undergoing rulemaking and my question is, is this going to be compatible with the current trend in the siting rulemaking.

MR. VERRELLI: That's our intent. There are other rules, for example, the things we have had on the street for what I call the interim degraded core that talks about vents. Obviously this requirement in 737 is identical with that and they would have to be consistent.

If you go to rulemaking on Part 100 of the source term you would just refer to that other position -- approved position.

MR. MOELLER: Okay, so you are trying to make sure that they are all compatible?

MR. VERRELLI: Yes, sir.

MR. MOELLER: On page 37 of my copy, which is vii, subparagraph (a), it's about inadequate core cooling. And I did not understand paragraph (a) as contrasted to paragraph (c). It says each boiling and pressurized lightwater nuclear power reactor licensee shall develop and implement procedures and training to be used by the operators to recognize the existence of inadequate core cooling and low coolant level in the reactor core using available instrumentation.

Now I could read that several ways: that he must be able to do this using the instrumentation that's there right now, that which is available; or he must install -- install adequate instrumentation so that in the future he can do this with what is available, because he's put is in.

And your paragraph (c) says, then, to put in the instruments to be able to do paragraph (a).

MR. VERRELLI: If he has to do (a), (b), and (c) before he gets the license he obviously must train a -- under (a) he must train wi h the instrumentation that is under (c). That was the philosophy.

MR. MOELLER: In my initial reading, as someone not wrapped up in the subject, if I read paragraph (c) first it would have helped me.

MR. BEARD: I would remind you also that the question of inadequate core cooling has been subject to some controversy as to whether or not what's installed to date in those plants is adequate or whether it needs to be supplemented.

MR. MOELLER: You're right.

MR. BEARD: So that thought is related to this.

MR. MOELLER: That's a very good point.

Lastly, it's not your problem, ut in some of the material that we were provided to study for the meeting we were reminded that the NRC Staff is supposed to be providing the ACRS with a quarterly report on rulemaking progress. I guess this is supposed to be a complete updating every quarter on the status of all rules and so forth.

Are we receiving these reports?

MR. VERRELLI: We have a representative who can speak to that.

MR. MOELLER: Let me quote. "The NRC Staff will provide the ACRS a status report on proposed rules on a quarterly basis", and I'd like to know when these are coming out and what they look like.

MR. CLEVELAND: There is a publication coming out each quarter updating the status of all regulations, either in process or proposed. It's published by the Office of Administration. I don't know whether you're actually receiving it or not, but I can't believe you're not.

MR. MOELLER: Well, I have seen perhaps one memo along these lines, but I just couldn't remember seeing them every quarter. Maybe we are.

MR. MATHIS: I can't answer the question. Rich, do you know?

MR. MAJOR: I'll find out for you.

MR. MATHIS: We probably have it someplace and just haven't looked at it.

MR. CLEVELAND: I will add I notice that this was just updated about one month ago. The current edition should be available about this time.

MR. MATHIS: Thank you.

MR. MOELLER: Thank you.

MR. EBERSOLE: Mr. Chairman, we seem to have gotten into the practice here of making a few comments.

MR. MATHIS: If it's substantive.

MR. EBERSOLE: On page 19 of the same document, as a matter of fact the same paragraph -- this is paragraph 11 -- a plan for collection of data shall be provided that will establish for ECCS systems and equipment outage dates, et cetera, et cetera.

All I want to comment on is that is pointed at ECCS systems and equipment -- emergency core cooling equipment. Those systems, as identified in PSARs and SARs and so forth, are peculiarly those after some sort of depressurization accident. They do not include AC/DC systems, water service systems, component cooling, all of the accessory systems which are actually having a higher demand than the ECS systems which are on duty to meet an occasional and hopefully rare challenge.

There are many systems out there which must work 24-hour-a-day for another year which may well not be interpreted to be ECCS systems, but they should in fact be put in a higher category than the ECCS systems. But they're not even mentioned.

MR. VERRELLI: Would you interpret diesels as an ECCS system?

MR. EBERSOLE: Yes, I would, or in a higher system which is not yet defined, which are completely performing emergency functions in a quiet manner, whose failure may be disasterous if the total function fails.

MR. VERRELLI: The use of the word "ECCS" may not be consistent with the intent of the requirement.

MR. EBERSOLE: And that they would perform -- such as constant moving of water and DC and AC power supplies, constant environmental control, things that go on when you are running the normal mode and m... continue to go on in the emergency mode and never cease.

MR. LAINAS: I think a general comment is there is too much emphasis on mitigation.

MR. BEARD: I think the intent of that particular comment was, sir, that in some cases the ECCS system, which is stand-by in nature -- say a valve fails and had to be repaired or had to be ordered and replaced, sometimes that can stretch out a little further than we may like for it to see. And we would like standby systems to be put back into an operable condition as soon as practicable.

So the point of this particular requirement was go back, Mr. Licensee or Mr. Applicant, and have a system that you can identify how often it's gone out, how long it's taken you to put it back and see if there is anything you can do to improve that.

MR. EBERSOLE: There should be a paragraph addressed to this.

MR. VERRELLI: Which would be a separate requirement.

MR. EBERSOLE: And a great deal more important.

MR. VERRELLI: One of the concerns that developed this request for that type of information was our whole method of permitting HPCI systems to be out for seven days and then you put it back in service. It could go back out and be another seven days.

Maybe we should change our whole approach to outages to be cumulative, as opposed to differential. So this was an attempt to obtain a certain type of data on outages of systems to see if we should revise our system.

MR. EBERSOLE: What if you put a battery on charge and you isolate the plant and you know perfectly well you don't need an accident. If that battery system fails you're going to have an accident.

MR. BEARD: I think we also have to bear in mind that I don't think anybody would disagree with your point that there are other areas beyond 737 where we need to make improvements.

MR. EBERSOLE: This bit about an output from TMI-2. We found that we need emergency output from heaters and we put that on the heaters, not remembering that they're not environmentally qualified. And, for that matter, the PORVs are not environmentally qualified for the hostile environment within which they sit to do their duty. But we put the half-page on that says get emergency power on and presumably that's the best we can do now without creating a convulsion in the industry.

You recall that BWRs have to -- over here it will be mentioned later on that you did require that for BWRs only. There's no valid reason there should be any difference. As a matter of fact, there might be a considerably better reason that the BWRs have better means to blow down because the boilers have about fifteen ways to do it. In the PWRs frequently you only have about two.

MR. MOELLER: Jesse, can I comment on that?

On page 24,490, in paragraph xliv in the center column at the bottom, in the center column it talks about the analysis shall be provided to demonstrate that for anticipated transients complicated by the worst single failure, and assuming proper operator actions, the core remains covered or no significant fuel damage results.

Then it's applicable to BWRs only. Why would that be? Is it all right for the PWRs to do this?

MR. EBERSOLE: Let me see that again.

MR. MOELLER: It's right here. There must be something in this that I don't see.

MR. EBERSOLE: That's a classic example of what I see.

MR. VERRELLI: The II.K items were developed as a result of independent evaluations by the Bulletins and

Orders task force on each of the venders. The PWRs, which was NUREG-626, was a report for Westinghouse and one for CER and one for B&W. This was their evaluation of the boilers after the TMI accident.

So having been identified, they are coupled with some experiences at Oyster Creek where they thought they had reached trouble with water level. We thought it was important to put it into NUREG-737. Thus, it appears here and why it did not necessarily appear as applicable to other plants.

You always have the question of identifying a problem and then saying how does it apply generically as opposed to a problem on a particular vendor. And a perfect example is Fort St. Vrain. Obviously you were involved in that. Are you adequately applying the lessons learned to something that is needed?

MR. EBERSOLE: There was something generated, by the way, that came out of Oak Ridge in March of this year about PWRs and their problems with the thermal shock which is followed by repressurization trnsients, and I notice as I read through here, there are numerous examples where you invite the operator to create that problem, whereas in this other area, you're inviting him he's better not create it.

So somewhere along the line we've got to rationalize the differences.

MR. VERRELLI: For the fixes you propose you are introducing a different problem. You have to be very careful, right.

MR. EBERSOLE: We're worried about high repressurization following a thermal transient and most of the language in here doesn't identify that as a problem. And, as a matter of fact, it didn't exist except way back in history.

MR. BEARD: I don't remember the number on it, but the thermal mechanical stress requirement is in there. I recognize, and you are correct, that with that lofty bureaucratic title it doesn't mean much. But when you read this requirement in the rule it references you to the NUREG item and there there are quite a few pages of information provided so it is clear.

MR. EBERSOLE: One other little point on xxxviii, the design of the system , steam line, pipe break protection circuitry shall be such that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent isolation of those systems. That's fine.

But I think really the predominant or certainly one of the significant causes of HPCI and RCIC isolation, has been the fact that the pipe galleries generate high temperatures in the state of need, the AC power system frequently fails, and the ambient temperatures detection circuity which was used for breaker detection malfunctioned, and you get isolation when you least need it.

MR. BEARD: As you recognize, one of the considerations has to be given to the fact that your safety system could be a source of the break, and there are provisions in there to detect potential breaks. High temperature is one of those.

Unfortunately, like lot of other things around here, you don't get something for nothing, so there are some drawbacks.

MR. EBERSOLE: When you have high temperature initiation, surely AC power failure will cause rising temperatures and shut the steam supply off when you precisely need it. So to permit that to be perpetuated by just this limited requirement here is not right.

MR. BEARD: I would remind you also there is another requirement that has to do with -- just below there, it's item xli, which I guess is 41, is just below the item you were talking about, which says the HPCI and RCIC systems shall be designed to withstand and operate satisfactorily following a complete loss of off-site power for at least two hours.

Now if you read the NUREG-737 description, it specifically points out to the reader that the major concern when that requirement was written was for a ventilation cooling in the space around that system.

MR. EBERSOLE: So it's covere., then, in more detail?

MR. VERRELLI: The first item you talked about, a lot of that outgrowth came from a program that has to -- they had inadvertent isolations and had to go in and fix that. So they wanted to go out to the other boilers and see if you have that same problem.

MR. MATHIS: Any other comments? Dade? Jerry?

W [illegible] I think we've covered this reasonably well and apparentl the draft is still being changed as it rolls along. And I gather that you are being responsive as you can to the comments you have received.

We will have a new draft out in two, maybe three, months, which we will be asked to look at at that particular time.

There's one other comment. I guess it seems to me that there is no question about this winding up as a rule. I think that decision has been made and there's no sense discussing that one any further.

MR. LAINAS: I think the Commission is going to take another shot at it.

MR. MATHIS: That's right, but it's going to be a rule. I think that's the target you are working towards.

MR. LAINAS: Exactly.

MR. MATHIS: So I guess from the Subcc⁻mittee's position, do we want to make any formal comment on this? Do you want to take it to the full Committee, or do we wait for the final rule?

Do you have any comment on that?

MR. RAY: I can comment. There's no question but that endorsement is going to be needed in its final form, if not before that. I just wonder if we would be justified in taking the time of the full Committee at the stage it's in now. I think the changes might be more appropriate to schedule it to go to the full Committee after the changes have been made in more or less final form.

You could report, for instance, at the next meeting that you have been over this quite thoroughly today and it was discussed in complete aspects and it looks like it's on course and we're satisfied with the progress we're making.

In fact, even if your schedule slips a month, as you indicated, I think you will have created a new track record that's unique within NRC activities and it very well could be held up as an example of what could be done.

MR. BEARD: We keep reminding the Power Division about that.

MR. EBERSOLE: If this goes to a rule I would like to make some qualifying statements when we bring this to the

Committee along the lines that I've generally mentioned here, that we've got lots of things in here which reflect on general matters that should be cleaned up in a better way than they are here.

Whether it's practical to do it and we've got to make a rule because somebody says we've got to make rules, all right. You've got to play the game. But that doesn't mean the game is what it ought to be.

MR. MATHIS: Well, I think when we get to that point, Jesse, we can remind people once again.

MR. RAY: You're saying another rule should be taken other than the rule changes, perhaps GDC changes?

MR. EBERSOLE: Whatever.

MR. MOELLER: It would seem to me that it would be very helpful, perhaps, to go beyond a Subcommittee Chairman report at the next meeting. If the full Committee is going to be reviewing this in February or January and writing on it, some sort of a preliminary orientation or a preliminary progress report might be in order.

I agree with Jesse there are going to be a number of Committee members that are going to have opinions on this. I don't mean to try to write anything, but if we could obtain an hour's time at the November -- that's probably too late, but, say, the December meeting just for a progress report by you and by the Staff.

MR. MATHIS: Well, I think that is definitely in order, but I think, again, it depends on when you're going to be ready. You have indicated here on your schedule that you will have a copy to us in December. Well, as we discussed earlier that may be a little optimistic.

MR. LAINAS: That may be optimistic. It's also predicated on the noticing. Now it's not out yet, so if the thing doesn't go out --

MR. BEARD: I guess I'm sort of sympathetic to what appears to be your desire to brief the larger group if they're going to write something on it.

MR. MOELLER: If you hit them cold in January or February without a little warning -- I mean, if nothing else it will call it to their attention and they'll begin to read it and be ready to respond.

MR. BEARD: Yes. The only hesitation I have in my own mind is that the status report in December may not be a whole lot different than the one we presented today

MR. MOELLER: That's all right with me. It doesn't bother me. But at least you alert the full Committee where you stand and where you're headed.

MR. VERRELLI: A summary of today's meeting is what you're saying?

MR. MOELLER: Yes.

MR. MATHIS: But I think many of the items we have discussed here today are obviously going to wind up being in the final draft and I think to do anything prior to the time we get that would be just premature, that's all.

MR. LAINAS: I think that many of the comments that we've noted are the same sort of concerns that we have and the same sort of concerns that were alerted in the rule itself, and we specified that these things would be under consideration and development.

We'll be glad to come back in December and tell you where we stand. I don't imagine the comments will be any different to the full Committee than the ones we heard today or the ones that we recognize. But we'll be happy to come back.

MR. RAY: This is a suggestion. Perhaps at the November meeting you could make a report on today's meeting and indicate the general character of what went on and ask the full Committee if they would like to have an interim report in December or January or something consistent with what this revised schedule would permit.

Therefrom you would get a pulse indication of just how they feel about it and they might also indicate the kind of thing they would like to have, which would help you to prepare for it.

MR. MATHIS: The kind of specific questions and things of that nature?

MR. RAY: Yes.

MR. MATHIS: Well, if there are no other comments on that, that's the route we will take.

As far as the specific date, I think we will let that dangle for the moment and we'll be in touch so we'll know when to plan for another session and see where we go from there.

MR. RAY: I would like to comment that the handout you made today with the various items in the rule cited at the top of the page and then the comments that were made and the responses you had is very useful, particularly for the next session.

MR. MATHIS: Yes, I would agree with that, Jerry.

Okay, do you have anything else, Gus?

MR. LAINAS: No, I don't. I want to say that I thank you for listening to us and we have noted your comments and we'll certainly take it into consideration when we redraft.

MR. MATHIS: We appreciate your taking the time to come down and explain to us, because I'm sure you can gather from our comments there have been some questions in our mind and we will take it from there.

And with that we will adjourn.

(Whereupon, at 10:25 o'clock a.m., the meeting was adjourned.)

NUCLEAR REGULATORY COMMISSION

This is to certify that the attached proceedings before the

____

in the matter of: ACRS/Subcommittee on TMI-2 Action Plans

Date of Proceeding: October 29, 1981

Docket Number: ____________

Place of Proceeding: Washington, D. C.

were held as herein appears, and that this is the original transcript thereof for the file of the Commission.

Ann Riley

Official Reporter (Typed)

Ann Riley

Official Reporter (Signature)

CONTACT: JTBEARD
X27465

ACRS BRIEFING

PROPOSED RULE FOR OL's
ON TMI ACTION PLAN

OCTOBER 29, 1981

- BACKGROUND
- CONTENT OF PROPOSED RULE
- RESPONSES
- PLANS AND SCHEDULE

BACKGROUND

• Commission Review of C.P. Rule; Request for O.L. Rule	3/10/81
• Proposed O.L. Rule Developed by Staff	4/17/81
• Commission Approval of Proposed Rule; Request for O.R. Rule	4/30/81
• Proposed O.L. Rule Published (46 FR 26491) (90-day comment period)	5/13/81
• Proposed O.R. Rule Developed by Staff	6/12/81
• Commission Decision on Proposed O.R. Rule - not to proceed	8/6/81
• Comment Period on Proposed O.L. Rule Ends	8/13/81

CONTENT OF RULE

A. GENERAL OBJECTIVE

- COMMISSION DETERMINED THAT NUREG-0737 REQUIREMENTS SHOULD BE CODIFIED INTO COMMISSION'S REGULATIONS
- NO INTENT TO CHANGE TECHNICAL CONTENT OF NUREG-0737

B. GENERAL FORMAT

- NEW PARAGRAPH (F) ADDED TO SECTION 50.34 "CONTENTS OF APPLICATION; TECHNICAL INFORMATION"
- TWO SUB-PARTS:

  (1) ITEMS TO BE IMPLEMENTED BY ISSUANCE OF OPERATING LICENSE (AFTER EFFECTIVE DATE OF RULE). (53 ITEMS)

  (2) ITEMS TO BE IMPLEMENTED EITHER BY O.L. ISSUANCE OR BY SPECIFIC DATES, WHICHEVER LATER. (I.E., "DATED REQUIREMENTS") (15 ITEMS)

C. ITEMS MERITING ADDITIONAL CONSIDERATION

DELETE CERTAIN TYPES OF ITEMS:

- SUFFICIENT INFORMATION RECEIVED; NO ADDITIONAL INFORMATION FROM APPLICANTS REQUIRED (E.G., BENCHMARK ANALYSIS OF SEQUENCE AFW FLOW)
- ITEMS MAY ALREADY BE SUFFICIENTLY CODIFIED (E.G., EMERGENCY PREPAREDNESS AND SUPPORT FACILITIES)
- POSITION IN -0737 MAY BE REVISED (E.G., OVERTIME LIMITS)
- RECONSIDERATION ON THE NEED FOR MODIFICATIONS (E.G., AUTO-TRIPPING OF RCPs)
- ITEMS ARE TOO DETAILED (E.G., FMEA ON ICS, CCI PORV)

D. TECHNICAL CONTENT

1. ITEMS TO BE IMPLEMENTED PRIOR TO ISSUANCE OF O.L. (THAT OCCURS AFTER EFFECTIVE DATE OF RULE)

   A. SHIFT MANNING

   B. TRAINING

   C. ADMINISTRATIVE CONTROLS

   D. PROCEDURE/DESIGN REVIEWS AND ANALYSES OF SB-LOCA

   E. PORV'S AND SAFETY VALVES

   F. PID CONTROLLER

   G. AUXILIARY FEEDWATER SYSTEMS

   H. EMERGENCY POWER SUPPLIES

   I. CONTAINMENT H_2 PENETRATION AND ISOLATION

   J. SAFETY INJECTION INDEPENDENT OF PRESSURIZER LEVEL

   K. ANTICIPATORY REACTOR TRIPS

   L. AUXILIARY HEAT REMOVAL SYSTEMS PROCEDURE

   M. REACTOR VESSEL LEVEL INDICATIONS

   N. FMEA ON ICS SYSTEM

   O. EFFECTS OF SLUG FLOW ON OTSG'S

   P. RCP SEAL DAMAGE ON LOSS OF SITE POWER

   Q. AUTO TRIP OF RCP'S

   R. HPCI AND RCIC SYSTEMS

   S. PRIMARY COOLANT LEAKAGE OUTSIDE CONTAINMENT

   T. AIRBORNE I_2 RADIATION MONITORING

   U. EMERGENCY PREPAREDNESS - FACILITIES

   V. NATURAL CIRCULATION ON REACTOR DEPRESSURIZATION

   W. ADS ACTUATION

2. ITEMS TO BE IMPLEMENTED BY DATE INDICATED OR BEFORE THE ISSUANCE OF O.L. --- WHICHEVER IS LATER

A. EMERGENCY PROCEDURES TO PREVENT INADEQUATE CORE COOLING (1/1/82)*

B. REACTOR COOLANT SYSTEM VENTS (7/1/82)

C. PLANT SHIELDING (1/1/82)*

D. POST ACCIDENT SAMPLING (1/1/82)*

E. RELIEF & SAFETY VALVE QUALIFICATION TESTING (7/1/82)

F. ACCIDENT MONITORING INSTRUMENTATION (1/1/82)*

G. INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (1/1/82)*

H. VOIDING IN RCS-ANALYSIS (1/1/82)*

I. ANALYSIS OF SEQUENTIAL AFW FLOW (1/1/82)*

J. AUTOMATIC PORV ISOLATION (1ST REFUELING OUTAGE 6 MOS. AFTER STAFF APPROVAL)

K. QUALIFICATION OF ADS ACCUMULATORS (1/1/82)*

L. PLANT SPECIFIC ANALYSIS OF SB-LOCA (1/1/83)

M. RCIC SUCTION AUTOMATIC TRANSFER (1/1/82)*

N. ADS AUTOMATIC ACTUATION (1ST REFUELING OUTAGE 6 MOS. AFTER STAFF APPROVAL)

O. CS AND LPCI REACTUATION (1ST REFUELING OUTAGE 6 MOS. AFTER STAFF APPROVAL)

*THESE ITEMS WILL MOVE TO SUB-PART (1) WHEN RULE IS PUBLISHED IN FINAL FORM (I.E., LATER THAN JANUARY 1, 1982)

RESPONSES

A. COMMENTS RECEIVED FROM:

• APPLICANTS, LICENSEES, OWNERS GROUPS	37
• NSSS VENDORS, ARCHITECT-ENGINEERS	6
• OTHERS (AIF, UCS, NRDC, CITIZENS)	6
TOTAL	49

B. GENERAL NATURE OF RESPONSES

- 39 GENERALLY OPPOSE RULEMAKING
  - LACK OF FLEXIBILITY
  - NEGATES PREVIOUS AGREEMENTS
  - DUPLICATES OTHER PARTS OF 10 CFR 50
  - TOO DETAILED
  - ITEMS RESOLVED
  - CRITERIA NOT FINALIZED BY NRC
- 4 GENERALLY FAVOR RULEMAKING
  - WILL MINIMIZE LITIGATION IN HEARINGS
  - WILL FURTHER SAFEGUARD PUBLIC SAFETY
- 5 NO GENERAL COMMENT EXPRESSED
  - OFFERED ITEM-SPECIFIC COMMENTS ONLY

C. SPECIFIC ITEMS RECEIVING MOST COMMENTS

(1)(i)	STAFFING	(60)
(1)(ii)	TRAINING	(36)
(1)(xliii)	SB-LOCA MODELS	(24)
(2)(vii)	INADEQUATE CORE COOLING	(24)
(1)(iv)	INDEPENDENT SAFETY EVALUATION GROUP	(23)
(1)(xvii)	CONTAINMENT ISOLATION	(17)
(1)(xii)	VALVE POSITION INDICATION	(16)
(1)(xlii)	REFERENCE WATER LEVEL	(16)
(2)(i)	EMERGENCY PROCEDURES	(16)
(1)(xxxii)	REPORTING VALVE CHALLENGES	(15)

PLANS AND SCHEDULE

• Supplemental Notice Published	October 81
- proposed O.R. rule not forthcoming	
- additional 30-day comment period	
• Extended Comment Period Ends	November 81
• Final O.L. Rule Developed by Staff (copy to ACRS)	December 81
• ACRS Review/Meeting on Final O.L. Rule	February 82
• Commission Decision on Final O.L. Rule	March 82

~~13. It is proposed to amend § 2429.23 by revising paragraph (d) to read as follows:~~

~~§ 2429.23 Extension; waiver.~~

~~* * * * *~~

~~(d) Time limits established in 5 U.S.C. 7117(c)(2) and 7122(b) may not be extended or waived under this section.~~

~~* * * * *~~

~~14. It is proposed to revise § 2429.25 to read as follows:~~

~~§ 2429.25 Number of copies.~~

~~Unless otherwise provided by the Authority or the General Counsel, or their designated representatives, as appropriate, or under this subchapter, any document or paper filed with the Authority, General Counsel, Administrative Law Judge, Regional Director, or Hearing Officer, as appropriate, under this subchapter, together with any enclosure filed therewith, shall be submitted in an original and four (4) copies. A clean copy capable of being used as an original for purposes such as further reproduction may be substituted for the original.~~

~~* * * * *~~

~~Note.—In accordance with section 605(b) of the Regulatory Flexibility Act of 1980, the Federal Labor Relations Authority and the General Counsel of the Federal Labor Relations Authority have determined that this document does not require preparation of a Regulatory Flexibility Analysis.~~

~~Dated: May 8, 1981.~~

~~Ronald W. Haughton,~~
~~*Chairman.*~~

~~Henry B. Frazier III,~~
~~*Member.*~~

~~Leon B. Applewhaite,~~
~~*Member.*~~

~~H. Stephan Gordon,~~
~~*General Counsel.*~~

~~[FR Doc. 81-14383 Filed 5-12-81; 8:45 am]~~

~~BILLING CODE 6727-01-M~~

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Licensing Requirements for Pending Operating License Applications

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission is proposing to add to its power reactor safety regulations a set of licensing requirements applicable to operating license applications. The requirements stem from the Commission's ongoing effort to apply the lessons learned from the accident at Three Mile Island to power plant licensing. Each applicant covered by the rule would have to meet these requirements, together with the existing regulations, in order to obtain an operating license.

DATE: Comments must be received on or before August 12, 1981.

ADDRESSES: Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

FOR FURTHER INFORMATION CONTACT: John A. Olshinski, Chief, Operating Reactors Assessment Branch, Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone 301-492-8069.

SUPPLEMENTARY INFORMATION: On March 28, 1979, the Three Mile Island Unit 2 (TMI-2) nuclear power plant experienced a loss of feedwater transient, complicated by a set of circumstances and events, culminating in the equivalent of a smallbreak loss-of-coolant accident with substantial core damage. The circumstances and events that caused the feedwater transient to develop into an accident include design deficiencies, equipment failures, and human errors.

In April 1979, the Commission established the Bulletin and Orders Task Force as the focal point for those TMI-2 related staff activities necessary to assure the immediate safety of all other operating power reactors. During May 1979, the efforts of this group resulted in the issuance of several IE Bulletins and Commission Orders covering a wide range of topics.

In May 1979, the Commission established the TMI-2 Lessons Learned Task Force to identify and evaluate safety concerns requiring prompt licensing actions for operating reactors (beyond the immediate actions taken as a result of the Bulletins and Orders Task Force effort) and for pending operating license applications. A set of short-term recommendations offered by this task force was published as NUREG-0578 in July 1979.

In addition to these special NRC task forces, several other official groups have investigated the accident at TMI-2 and developed recommendations. These groups include the Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC's staff's Task Force on Emergency Planning, and the NRC Office of Standards Development and Nuclear Regulatory Research. Each of the investigating groups, acting independently, organized their recommendations in a different way. A steering group was appointed to organize and assess the many recommendations and to develop the "TMI-2 Action Plan", which would provide a comprehensive and integrated plan for all actions necessary to correct or improve the regulation and operation of nuclear facilities. The items identified by the Lessons Learned Task Force and many longer term generic items identified by the Bulletins and Orders Task Force were included in the Action Plan program. This Action Plan was published as NUREG-0660 in May 1980.

In reviewing the technical, schedular and cost aspects of the numerous items of the TMI Action Plan, the Commission has approved a number of actions that provide substantial additional protection which is required for public health and safety. The Commission asked the staff to obtain industry comments on the approved Action Plan items and to make appropriate revisions prior to finalizing the requirements.

Actions to improve the safety of nuclear power plants now operating were judged to be necessary immediately after the accident and could not be delayed until the Action Plan was developed, although they were subsequently included in the Action Plan. Before these immediate actions were applied to operating plants, they were approved by the Commission. Many of the required immediate actions have already been taken by licensees and most are scheduled to be completed in the near future.

On May 15, 1980, after review of the last version of the Action Plan, the Commission approved a list of "Requirements for New Operating Licenses", contained in NUREG-0694. On October 28, 1980, the Commission approved a "Clarification of TMI Action Plan Requirements", now contained in NUREG-0737, which supersedes NUREG-0694. On December 16, 1980, the Commission issued a statement of policy, "Further Commission Guidance for Power Reactor Operating Licenses", which replaced a previous policy statement issued on June 16, 1980.

On September 5, 1980, the NRC sent letters regarding the new requirements approved by the Commission in its consideration of the TMI Action Plan to all licensees of operating reactors, applicants for operating licenses, and holders of construction permits. During the week of September 22, 1980 regional meetings were held to provide more detailed explanation of the new requirements and to obtain industry comments. Based upon these discussions, the finalized Action Plan requirements were issued on October 31, 1980, as NUREG–0737, which included a summarizing letter. The letter noted that NUREG–0737 includes in tabular form and with technical clarification all the post-TMI-2 requirements that had been approved at that time by the Commission, but does not constitute the totality of the TMI-2 Action Plan.

Since NUREG–0737 was issued, the Commission has determined that the new requirements should be codified into the Commission's regulations. While there is no intent to change the technical content of these requirements, the NUREG–0737 items have been re-written in language appropriate for the Commission's regulations.

Substance of the Rule

This rule, which addresses the same set of items contained in NUREG–0737, imposes new safety requirements for operating license applications. The Commission has determined that these requirements must be met by all applicants for operating licenses. It should be noted, however, that there are many elements in the TMI Action Plan (NUREG–0660) not included in NUREG–0737, that have not yet been developed by the staff or acted upon by the Commission. There are also items that the Commission has directed to be the subject of further study. This rule will be augmented in the future to add new requirements as they are approved. Opportunity for public comment will be provided when such additional requirements are contemplated.

For the sake of completeness, all of the basis requirements of NUREG–0737 are incorporated in this proposed rule. It is recognized that some of the items individually are or may be the subject of other rulemakings (e.g., shift manning, operator qualification, and interim degraded core cooling). The Commission does not intend to issue duplicative rules. Consolidation or other appropriate action will be taken before final rulemaking to be sure that each subject is addressed in only one place in the rules.

While this rule contains the basic requirements set out in NUREG–0737, it does not incorporate the entirety of the document. In particular, the rule does not contain the detailed criteria, staff positions, and guidance contained in NUREG–0737 for satisfying many of the requirements. To have included such detail would have resulted in an excessively detailed and restrictive rule. However, the Commission has reviewed NUREG–0737 and has concluded that the positions contained therein provide a basis for responding to the experience of the TMI-2 accident. Applicants may, of course, propose to satisfy the rule's requirements by a method other than that detailed in NUREG–0737, but in such cases must provide a basis for determining that the requirements of the rule have been met.

In developing this proposed rule, the Commission has recognized that there are a number of items from NUREG–0737 that merit additional consideration prior to being included in a final rule. For example, there are several items for which the ongoing Commission review, based on submittals by operating reactors, may resolve the concern such that no additional information on these items would be needed for operating license applications. Some items may be redundant with existing regulations. Some items are presently under Commission review with preliminary indications that either the requirement may not be needed or the specific criteria in NUREG–0737 for meeting the requirement may be revised. Finally, some items are so specific and of limited applicability that their inclusion in the regulations may not be warranted. Accordingly, while the proposed rule presently contains all items from NUREG–0737 applicable to operating license applications, comment is specifically solicited on items that may not need to be included for the reasons discussed above. The following are examples of items that have been identified as candidates for such reconsideration.

(a) Generic items for which sufficient information may have already been received and no additional information may be needed from OL applicants:

II.K.2.15 Effects of Slug Flow on OTSG Tubes.
II.K.2.17 Voiding in RCS (complete for B&W only).
II.K.2.19 Benchmark analysis in Sequential AFW Flow to the OTSG.

(b) Items that may already be sufficiently codified in the regulations:

II.K.3.30 Upgrade of SBLOCA Model.
II.K.2.31 Plant Specific Analysis to Show Conformance with 10 CFR 50.46.
III.A.1.2 Upgraded Emergency Support Facilities.
III.A.2 Emergency Preparedness—Long Term.
III.D.3.4 Control Room Habitability.

(c) Items for which the Commission position on acceptability in NUREG–0737 may be revised:

I.A.1.3 Overtime Limitations.
I.C.6 Verify correct performance of Operating Activities.

(d) Items that are presently under Commission staff review and reconsideration as to whether the modifications are needed:

II.K.3.5 Automatic RCP Trip for PWRs.
II.K.1.21 Anticipatory Trip on LOFW, Turbine Trip and Low S/G level (B&W).
II.K.2.10 Same as II.K.1.21.
II.K.1.20 Procedures for Manual Trip on Specific Events (B&W).

(e) Items that may be too detailed or of limited applicability to merit codifying in the regulations:

II.K.2.2 Initiation and Control of AFW Independent of ICS (B&W)
II.K.2.9 FMEA of the ICS (B&W)
II.K.3.9 Modifications to the PID Controller for W-designed Plants
II.K.3.10 Anticipatory Reactor Trip Bypass Setpoint
II.K.3.11 PORVs Manufactured by CCI, Inc.

The proposed rule includes a provision that the Commission may, for good cause shown, grant relief from the required implementation schedules on a case-by-case basis. The Commission recognizes that this rule may affect operating license proceedings now pending before the adjudicatory boards. While this may often be true when a new rule is promulgated, the broad scope and relative detail of this rule could cause a greater than usual impact on pending proceedings. This impact might be particularly severe on proceedings where the record has been closed. The Commission solicits comments on the potential impact of this rule, and its implementation schedule, on pending operating license proceedings.

Based upon its extensive review and reconsideration of the issues arising as a result of the Three Mile Island accident, the Commission has decided that applications for an operating license should be measured by the NRC staff and Presiding Officers in adjudicatory proceedings against the existing regulations, as augmented by this rule. It is the Commission's view that this new rule, together with the existing regulations, form a set of regulations, conformance with which meets the requirements of the Commission for issuance of an operating license. The Commission seeks public comment on the requirements contained in this rule. It should be noted that the Commission intends to augment its regulations with a similar rule for operating reactors. The proposed Operating Reactor Rule will be published for comment within two to three weeks of the publication of this proposed rule for Operating License Applications.

Paperwork Reduction Act

The proposed rule will be submitted to the Office of Management and Budget for clearance of the application requirements that may be appropriate under the Paperwork Reduction Act (Pub. L. 96-511). The SF-83 "Request for Clearance," Supporting Statement, and related documentation submitted to OMB will be placed in the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555. The material will be available for inspection and copying for a fee. No license may be issued unless a completed application form has been received that meets the requirements of 10 CFR 50.34(f). (42 U.S.C. 2201, 5841, 5844).

Regulatory Flexibility Act

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and Section 552 and 553 of Title 5 of the United States Code, the Commission proposes to amend Part 50 of Chapter I, Title 10 of the Code of Federal Regulations as follows:

The authority citation for Part 50 reads as follows:

Authority: Secs. 103, 104, 161, 182, 183, 68 Stat. 936, 937, 948, 953, 954, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 (42 U.S.C. 5842, 5846), unless otherwise noted. Section 50.78** also issued under sec. 122, 68 Stat. 939, 42 U.S.C. 2152. Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended, 42 U.S.C. 2234. Sections 50.100–50.102 issued under sec. 186, 68 Stat. 955; 42 U.S.C. 2236. For the purposes of sec. 223, 68 Stat. 958, as amended; 42 U.S.C. 2273, § 50.54(i) issued under sec. 161i, 68 Stat. 949; 42 U.S.C. 2201(i), and §§ 50.70–50.71 and § 50.78** issued under sec. 161o, 68 Stat. 950, as amended; 42 U.S.C. 2201(o) and the Laws referred to in Appendices.

1. A new paragraph (f) is added to § 50.34 to read as follows:

§ 50.34 Contents of application, technical information.

* * * * *

(f) *Additional TMI-related requirements for applications for an operating license.* In addition to the requirements of paragraph (b) of this section, each application for an operating license that is to be issued after (* * * insert effective date of this rule * * *) shall meet the requirements in paragraphs (f) (1) and (2) of this section. If the applicant contends that implementation of an item on the schedule set forth in this rule is impractical for its facility, the applicant may provide information to support this contention. The Commission will evaluate this information and, based on its determination of earnest effort and good cause shown, may grant relief from the implementation schedule, on a case-by-case basis. In such cases, the Commission will impose alternative schedule requirements suitable for that facility.

(1) For the following requirements, the application shall describe how each requirement will be implemented or satisfied prior to issuance of an operating license.

(i) The minimum shift staffing for operators, licensed and non-licensed, shall be as shown in Table 1. In addition to the staffing requirements stated in the Table, each operating shift, except during periods of cold shutdown, shall include a qualified Shift Technical Advisor (STA). In addition to the staffing requirements stated above, shift crew assignments shall include a licensed senior reactor operator to directly supervise core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties. The amount of overtime worked by plant staff members performing safety-related functions shall be limited. Other onshift staffing and emergency response capabilities shall be as shown in Table 2. The capability for augmentation of resources for emergency response functions shall be equivalent to that shown in Table 2. (I.A.1.1;[1] I.A.1.3; III.A.1.2)

(ii) The operator initial training and requalification programs shall include: heat transfer, fluid flow, and thermodynamics; and emphasis on reactor and plant transients.

[1] Alphanumeric designations correspond to the related action plan items in NUREG-0737, "Clarification of the TMI Action Plan Requirements" and NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." They are provided herein for information only.

Table 1.—*Required Shift Manning*

Operating status	One unit, one control room	Two units, one control room	Two units, two control rooms	Three units, two control rooms
One Unit Operating*	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
	1 SRO	1 SRO	1 SRO	1 SRO
	2 RO	3 RO	3 RO	4 RO
	2 AO	3 AO	3 AO	4 AO
Two Units Operating*	NA	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
		1 SRO	2 SRO	[1] 2 SRO
		3 RO	4 RO	[1] 5 RO
		3 AO	4 AO	
				5 AO
All Units Operating*	NA	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
		1 SRO	2 SRO	2 SRO
		3 RO	4 RO	5 RO
		3 AO	4 AO	5 AO
All Units Shut Down	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)	1 SS (SRO)
	1 RO	2 RO	2 RO	3 RO
	1 AO	3 AO	3 AO	5 AO

SS—shift supervisor.
SRO—licensed senior reactor operator.
RO—licensed reactor operator.
AO—auxiliary operator.
[1] Only 1 SRO and 4 ROs required if both units are operated from one control room.
NOTE.—(1) In order to operate or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each such unit.
(2) In addition to the staffing requirements indicated in the table, a licensed senior operator will be required to direct/supervise any core alteration activity.
(3) See item I.A.1.1 for shift technical advisor requirements.
*Modes 1 through 4 for PWRs.
Modes 1 through 3 for BWRs.

Table 2.—*Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies*

Major functional area and location	Major tasks	Position title or expertise	On shift*	Capability for additions 30 min	Capability for additions 60 min
Plant Operations and Assessment of Operational Aspects		Shift Supervisor (SRO)	1		
		Shift Foreman (SRO)	1		
		Control Room Operators	2		
		Auxiliary Operators	2		
Emergency Direction and Control (Emergency Coordinator)***		Shift Technical Advisor, Shift Supervisor or designated facility manager	1**		
Notification Communication****	Notify licensee, State, local and Federal personnel and maintain communication		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency Operations	Senior Manager			1
	Facility (EOF) Director				
	Offsite Dose	Senior Health Physics			
	Assessment	(HP) Expertise		1	
	Offsite Surveys			2	2
	Onsite (out-of-plant)			1	1
	In-plant surveys	HP Technicians	1	1	1
	Chemistry/Radio chemistry	Rad/Chem Technicians	1		1
Plant System Engineering, Repair and Corrective Actions	Technical Support	Shift Technical Advisor	1		
	Core/Thermal Hydraulics			1	
	Electrical				1
	Mechanical				1
	Repair and Corrective Actions	Mechanical Maintenance/Rad Waste Operator	1**		1
		Electrical Maintenance/Instrument and Control	1**	1	1
		(I&C) Technician		1	
Protective Actions (In-Plant)	Radiation Protection	HP Technicians	2**	2	2
	a. Access Control				
	b. HP Coverage for repair, corrective actions, search and rescue first aid and firefighting				
	c. Personnel monitoring				
	d. Dosimetry				
Firefighting			[1]	[3]	[3]
Rescue Operations and First-Aid			2**	[3]	[3]
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	[2]		
		Total	10	11	15

[1] Fire Brigade per Technical Specifications.
[2] All per security plan.
[3] Local support.

NOTES.
*For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.
**May be provided by shift personnel assigned other functions.
***Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.
****May be performed by engineering aide to shift supervisor.

The training program for all operating personnel shall include training to recognize, control and mitigate the consequences of accidents in which the core is severely damaged. In addition, each applicant shall support the development of its training program, emergency procedures and control room hardware, with applicable human engineering data. Additionally, intensive and comprehensive training exercises are to be conducted during low-power testing programs to provide experience for each operating shift. The principal instructors shall be qualified at the senior reactor operator level and shall periodically thereafter demonstrate their continued competency. An applicant for a senior reactor operator license shall have had experience as an operator and shall participate in an NRC approved training program. In addition to the written examination and the oral examination administered in the plant, and operational examinations on an appropriate simulator will be administered by the NRC. The minimum passing grade shall be 80% overall with a minimum in each technical category of 70%. (I.A.2.1; I.A.2.3; I.A.3.1; I.G.1; II.B.4)

(iii) Corporate management directives shall be issued that emphasize the shift supervisor's role in the control room as the primary onsite manager responsible for safe operation of the plant under all conditions. Such directives shall clearly define his responsibilities and authority including his command decision authority, relative to other plant management personnel, over plant operations personnel. The shift supervisor's responsibilities shall include limiting personnel access to the control room during emergencies; his administration duties shall be such that they do not detract from or are subordinate to the management responsibility for assuring the safety operation of the plant. Training programs for shift supervisors shall strengthen both management and operational capabilities (I.A.1.2, I.C.3, I.C.4)

(iv) An onsite independent safety engineering group of technically qualified personnel shall be provided to perform continuing systematic reviews of plant activities, including operating experience information that may indicate areas for improving plant safety. This group shall also provide recommendations and advice to an offsite high level corporate technical officer, not in the management chain for power production. (I.B.1.2)

(v) Analyses of small-break loss-of-coolant accidents and of transients and accidents that involve postulated multiple failures, consequential failures, and operator errors, which if unmitigated could lead to inadequate core cooling, shall be provided. The analyses shall be carried sufficiently into the event to identify all significant thermal/hydraulic/neutronic phenomena and to address possible failures and operator errors during the long-term cooling phase. Emergency procedure guidelines to mitigate these transients and accidents shall be provided. (I.C.1)

(vi) Administrative controls shall be provided to ensure adequate exchange of plant status information between control room operations personnel during shift and relief turnover. As a minimum, the exchanged information shall include: Values of key plant parameters, availability and alignment of systems important to safety, identification of systems and components in an acceptable degraded mode of operation, and identification of systems out of service for maintenance or test. (I.C.2)

(vii) A management system shall be provided to perform the following functions: (A) Review operating experience information originating both within and outside the facility; (B) Promptly supply information pertinent to plant safety, including proposed procedural changes and plant modifications, to operators and other appropriate plant personnel; and (C) Assure that such information is incorporated into training and requalification programs. (I.C.5)

(viii) A management system shall be provided to independently verify the proper performance of operational and maintenance activities, as a means of reducing errors that could result in or contribute to accidents. The system shall include automatic status monitoring or verification by two qualified individuals (I.C.6)

(ix) Reviews of the proposed procedures for low-power test, power ascension tests, and emergency procedures to verify the adeqacy of procedures shall be obtained from the nuclear steam system supplier. (I.C.7)

(x) Detailed reviews of the final design shall be performed to ensure that the design of the control room and control boards are in conformance with good human factors engineering principles and that information for the control room operators is presented in a manner that facilitates recognition of developing off-normal conditions, and mitigation of accidents. (I.D.1)

(xi) A plan for collection of data shall be provided that will establish for ECCS systems and equipment: (A) Outage dates and durations, (B) cause of the outage, (C) systems or components involved, (D) specific corrective actions taken, and (E) changes that may improve ECCS equipment availability. (II.K.3.17)

(xii) Direct position indications (open or closed) for the relief and safety valves shall be provided in the control room. (II.D.3)

(xiii) The auxiliary feedwater system (AFW) shall be evaluated including: (A) A simplified AFW reliability analysis using event-free and fault-free logic techniques; (B) A design review of AFW; and (C) An evaluation of AFW flow design bases and criteria. (Applicable to PWRs only). (II.E.1.1)

(xiv) The protection system shall include automatic and manual initiation of the auxiliary feedwater system and control room indication of system flow. (Applicable to PWRs only). (II.E.1.2)

(xv) The design shall include the capability to promptly connect onsite electric power to: (A) Pressurizer heater and associated controls sufficient to establish and maintain natural circulation in hot standby conditions, (B) pressurizer power-operated relief valves, (C) the block valves for the pressurizer power-operated relief valves, and (D) pressurizer water level instrumentation. (Applicable to PWRs only). (II.E.3.1 & II.G.1)

(xvi) Each power reactor that relies upon external recombiners or purge repressurization systems to satisfy the requirements of § 50.44 of this part shall be provided with containment penetrations for the external recombiners or purge/repressurization systems that either: (A) Are dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures and are sized to satisfy the flow requirements of the external recombiners or purge/repressurization systems, or (B) are of a combined design for use by either external recombiners or purge/repressurization systems and other systems, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/pressurization systems, and are sized to satisfy the flow requirements of the external recombiners or purge/pressurization systems. (II.E4.1)

(xvii) The containment isolation system design shall provide that: (A) All non-essential systems are isolated automatically, (B) each non-essential penetration (except instrument lines), has two isolation barriers in series, (C) the overriding (resetting) of the isolation signal shall require deliberate operator actions of at least two steps and no single sequence of operator override actions shall cause the reopening of the containment penetrations associated with more than one system or more than one purge or vent isolation valve, (D) the containment high pressure set point for initiating containment isolation is as low as is compatible with normal operation, and (E) all containment purge and vent isolation valves will receive an automatic closure signal on containment high radiation. (II.E.4.2)

(xviii) A review shall be provided of all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper Engineered Safety Features (ESF) functioning. (II.K.1.5)

(xix) Procedures for removing safety-related systems from service (and restoring to service) shall be provided that ensure that operability status will always be known by the control room operators. (II.K.1.10)

(xx) Safety injection shall be initiated when the pressurizer low pressure setpoint is reached regardless of the pressurizer level. (Applicable to Westinghouse-designed reactors only.) (II.K.1.17)

(xxi) The reactor protection system shall include anticipatory reactor trip for loss of main feedwater, turbine trip, and significant decrease in steam generator level. Procedures and associated operator training shall be provided to ensure prompt manual reactor trip for main steamline isolation valve closure, loss of offsite power, and low pressurizer level. (Applicable to Babcock & Wilcox-designed reactors only.) (II.K.1.20, II.K.1.21, and II.K.2.10)

(xxii) An analysis shall be provided to verify that the power-operated relief valves on the pressurizer will open during less than five percent of all anticipated overpressure transients for the range of plant conditions which might occur during a fuel cycle. (Applicable to Babcock-Wilcox-designed reactors only.) (II.K.2.14 and II.K.3.7)

(xxiii) The design of the auxiliary heat removal systems shall be such that necessary automatic actions will occur, and manual actions can be taken, when the main feedwater system is not operable. (Applicable to BWRs only.) (II.K.1.22)

(xxiv) A description shall be provided of all reactor vessel level indications used for automatic or manual initiation of safety systems. Other instrumentation that might give the operator the same information on plant status shall also be described. (Applicable to BWRs only.) (II.K.1.23)

(xxv) Procedures and training shall be provided for operating personnel relative to initiation and control of auxiliary feedwater independent of the Integrated Control System. (Applicable to Babcock & Wilcox-designed reactors only) (II.K.2.2)

(xxvi) A failure modes and effects analysis of the integrated control system (ICS) shall be provided. (Applicable to Babcock and Wilcox-designed reactors only.) (II.K.2.9)

(xxvii) A detailed analysis of thermal-hydraulics conditions in the reactor vessel during recovery from a small-break LOCA, with an extended loss of all feedwater, requiring the use of the cooler high-pressure injection system water, shall be provided to confirm that vessel integrity is not jeopardized. (Applicable to PWRs only.) (II.K.2.13)

(xxviii) An analysis shall be provided of the effects of slug flow on the once-through steam generator tubes after primary system voiding. (Applicable to Babcock & Wilcox-designed plants only.) (II.k.2.15)

(xxix) An evaluation shall be provided of the potential for and impact of reactor coolant pump seal damage and leakage upon loss of offsite power. If such damage is indicated, an analysis shall be provided of the limiting small-break loss-of-coolant accident complicated by subsequent reactor coolant pump seal damage. (II.k.2.16 and II.k.3.25)

(xxx) For Westinghouse-designed facilities where the reactor trip is to be bypassed when operating below 50 percent power, an evaluation shall be provided to verify that the probability of a small break LOCA resulting from a stuck-open PORV is not significantly greater than the case where this trip is bypassed only when operating below 10 percent power. (II.k.3.10)

(xxxi) An analysis shall be provided that defines the probability of a small-break LOCA caused by a stuck-open power operated relief valve (PORV). If this probability is a significant contributor to small-break LOCAs from all causes, provide a design description for an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.) (II.k.3.2 and II.k.3.1)

(xxxii) Any failure of a safety or relief valve shall be reported promptly to the NRC and all challenges to such valves shall be reported annually. (Applicable to PWRs only.) (II.k.3.3)

(xxxiii) An evaluation shall be provided of the automatic tripping of the reactor coolant pumps in the case of a small-break loss-of-coolant accident. (Applicable to PWRs only.) (II.k.3.5)

(xxxiv) If a proportional integeral-derivative controller is installed in the power operated relief valve (PORV) control system, the control system shall be operated so as to preclude opening the PORV due to derivate action. (Applicable to Westinghouse-designed reactors only.) (II.k.3.9)

(xxxv) Complete justification shall be provided for the use of any type of pressure-operated relief valve that has failed during testing (such as those supplied by Control Components, Inc., that failed during hot functional testing at a plant). (Applicable to PWRs only.) (II.k.3.11)

(xxxvi) An anticipatory reactor-trip on turbine-trip shall be provided. (Applicable to Westinghouse-designed reactors only.) (II.k.3.12)

(xxxvii) An evaluation shall be provided of the safety effectiveness of initiating the reactor core isolation cooling system at a higher water level than that for the high pressure coolant injection system and of restarting both systems on low water level. (Applicant to BWRs only.) (II.k.3.13)

(xxxviii) The design of the HPCI/RCIC steam line pipe-break detection circuitry shall be such that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent isolation of these systems. (Applicable to BWRs only.) (II.k.3.15)

(xxxix) An analysis shall be provided to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems, shall be provided. (Applicable to BWRs only.) (II.k.3.16)

(xl) Pending the implementation of automatic transfer features, clear and cogent procedures shall exist for manual transfer of RCIC system suction to the suppression pool when the condensate storage tank level is low. (Applicable to BWRs only) (II.k.3.22)

(xli) The HPCI and RCIC systems shall be designed to withstand and operate satisfactorily following a complete loss of offsite power for at least two hours. (Applicable to BWRs only.) (II.K.3.24)

(xlii) The scales of the various reactor vessel water level instruments shall be referenced to the same point. (Applicable to BWRs only.) (II.K.3.27)

(xliii) Small-break loss-of-coolant accident analysis methods used to comply with Appendix K to 10 CFR Part 50 shall be revised and provided that account for experimental data, including data from the Loss-of-fluid-test (LOFT) and Semiscale Test facilities. This evaluation shall consider the LOFT test, (L3-6). (II.K.3.30)

(xliv) Analysis shall be provided to demonstrate that for anticipated transients complicated by the worst single failure, and assuming proper operator actions, the core remains covered or no significant fuel damage results from core uncovery. (Applicable to BWRs only). (II.K.3.44)

(xlv) Analysis shall be provided to support depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWRs only.) (II.K.3.45)

(xlvi) Each boiling and pressurized light-water nuclear power reactor applicant shall implement leak reduction measures to that leakage, from systems outside containment (systems that would or could contain highly radioactive fluids during and following a serious transient or accident), is eliminated or minimized to the maximum extent practicable to prevent the release of significant amounts of radioactive material during and following an accident. Consideration shall be given to reductions of potential release paths that could result from design or operator deficiencies.

(xlvii) Each boiling and pressurized light-water nuclear power reactor licensee shall establish and implement a program of preventive maintenance to eliminate or minimize, to the maximum extent practicable, leakage from systems outside containment. This program shall include periodic (integrated) leak tests of these systems at intervals not to exceed each refueling cycle and also include (as-well-as) the reduction of potential release paths by appropriate operator training. (III.D.1.1)

(xlviii) Each boiling and pressurized light-water power reactor shall be provided with instrumentation, equipment and associated training and procedures for determining, under accident conditions, the airborne radioiodine concentration in areas within the facility where plant personnel may be present during and following an accident. (III.D.3.3)

(xlix) The control room and associated habitability systems shall be designed to adequately protect the reactor operations staff against the effects of accidental release of toxic or radioactive gases such that the nuclear plant can be operated or safety shutdown under accident conditions. Analysis based upon the final as-built conditions shall be provided to demonstrate that airborne concentrations of such hazardous fumes will permit control room operators to remain in the control room to take appropriate safety actions. (III.D.3.4)

(l) Dedicated emergency response facilities shall be established and maintained for command and control, support, and coordination of onsite and offsite functions during reactor accident conditions. The Technical Support Center is to provide an appropriate near-the-control-room location for those individuals who are knowledgable of and responsible for engineering and management support of reactor operations, to diagnose and evaluate plant conditions and for more orderly conduct of plant activities during emergency conditions. The Operational Support Center is to provide an area seperate from the control room for shift and other support personnel (e.g., auxiliary operator, technicians, health physics personnel) to report for instructions from the control room staff. The near-site Emergency Operations facility is to provide (A) a center for analysis of plant effluents, meteorological conditions, offsite radiation measurements and for offsite dose projections, and (B) a center for coordination of all licensee onsite and offsite activities and coordination with Federal, State, and local authorities for implementation of offsite emergency plans.

(ii) Plans and facilities for coping with emergencies shall be in accordance with the requirements set forth in other sections of 10 CFR Part 50. (III.A.1.1.; III.A.1.2; III.A.2)

(iii) The design shall ensure the capability of natural circulation in the event that depressurization of the reactor vessel, during a small break LOCA, is required (II.K.3.46)

(iiii) A feasibility study and risk assessment shall be submitted that defines the optimal approach for eliminating the need for manual actuation of the automatic depressurization system to assure adequate core cooling. (Applicable to BWRs only) (II.K.3.18)

(2) These requirements shall be implemented either by the date indicated or before the issuance of an Operating License, whichever is later. The application shall describe how each requirement will be implemented or satisfied.

(i) Emergency procedures shall be provided to mitigate small-break loss-of-coolant accidents, and transients and accidents that involve postulated multiple failures, consequential failures, and an operator errors, which, if unmitigated, could lead to inadequate core cooling. (January 1, 1982) (I.C.1)

(ii) Each boiling and pressurized light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system and reactor vessel head and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause their loss of function, remotely operated from the control room, to provide improved operational capability to maintain adequate core cooling following an accident. High point vents are not required, however, for the tubes in U-tube steam generators. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B to this Part 50. In particular, the vent system shall be designed to ensure a low probability that (A) the vents will not perform their safety functions and (B) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident. (July 1, 1982) (II.B.1)

(iii) Each boiling and pressurized light-water nuclear power reactor shall be provided with both adequate access to areas that may be used during and following an accident and protection of safety equipment so that an accident that results in the release of large amounts of radioactive material will not limit personnel occupancy or degrade safety equipment by the radioaction fields that may exist during and following the accident to the extent that required safety functions cannot be accomplished.

(A) The facility design must be based on a release of radioactive material from the fuel to the primary coolant system that is not less than 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of the remaining core fission products. For equipment and areas affected by the reactor coolant, it shall be assumed that the above distribution of radioactive material is intimately mixed with the coolant water except that recirculated, depressurized coolant water may be assumed to contain no noble gases. For equipment and areas affected by the containment atmosphere, it shall be assumed that not less than 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen inventory are uniformly dispersed in the containment atmosphere and an additional 25% of the core equilibrium halogen inventory and 1% of the remaining core fission products are uniformly distributed on surfaces exposed to the containment atmosphere.

(B) The facility design basis must be such that an individual operator will not receive more than a 5 rem whole body dose, or its equivalent to any part of the body, while performing a necessary safety function during and following an accident. (January 1, 1982) (II.B.2)

(iv) Each boiling and pressurized light-water nuclear power reactor shall be provided with the capability for personnel to obtain and quantitatively analyze a reactor coolant or containment atmosphere sample during and following an accident.

(A) The facility design must be based on the radioactive material release terms described in paragraph (*f*)(1)(iii) of this section.

(B) The design basis for the plant equipment that provides the capability to obtain and analyze a sample must be based on the assumption that it will be done promptly, and without incurring a radiation exposure to any individual in excess of 5 rem to the whole body, or its equivalent to any part of the body.

(C) The capability to quantitatively analyze a sample must be based on the use of either in-line monitoring or an onsite radiological and chemical analysis facility. If in-line monitoring is chosen, a capability must be provided for backup sampling using grab samples, and must include the capability for analyzing the samples at either an onsite or offsite facility. The analysis capability must provide, as needed, quantification of the following:

(*1*) Those radioisotopes necessary to indicate the extent of core damage;

(*2*) Hydrogen in the containment atmosphere;

(*3*) Total dissolved gases or dissolved hydrogen gas in the reactor coolant;

(*4*) Boron in the reactor coolant; and

(*5*) Chloride in the reactor coolant. Chloride analyses may be performed offsite and are not required to be done promptly. (January 1, 1982) (II.B.3)

(v) Qualification tests shall be conducted on the reactor coolant system relief and safety valves and, for PWRs, block valves, for all fluid conditions under operating conditions, transients and accidents. Block valves for each relief valve shall be qualified to isolate not only a leaking relief valve under normal conditions, but also any fluid flow conditions generated by a stuck-open relief valve under normal operating or accident conditions. The results of the qualification tests shall be submitted. (Applicable to PWRs only) (July 1, 1982) (II.D.1)

(vi) Accident Monitoring Instrumentation shall be provided for each boiling and pressurized light-water nuclear power reactor and shall have the capability during and following an accident for:

(A) Providing and recording in the control room a continuous indication of:

(*1*) Containment pressure;

(2) Hydrogen concentration in the containment atmosphere;

(3) Containment water level;

(4) Containment radiation level; and

(5) Radioactive noble gas concentrations in the plant gaseous effluents at all potential accident release paths effective.

(B) Quantifying the concentration of radioiodines and radioactive particulates in plant gaseous effluents at all potential accident release paths.

(C) Performing their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section. (January 1, 1982) (II.F.1)

(vii) (A) Each boiling and pressurized light-water nuclear power reactor licensee shall develop and implement procedures and training to be used by the operators to recognize the existence of inadequate core cooling and low coolant level in the reactor core using available instrumentation.

(B) Each pressurized light-water nuclear power reactor shall be provided with a primary coolant saturation meter (subcooling meter) that provides in the control room a continuous, recorded, on-line indication of the primary coolant saturation condition.

(C) Each boiling and pressurized light-water nuclear power reactor shall be provided with an instrumentation system, for example, reactor vessel water level indicators for pressurized water reactors that augment the incore thermocouples; and incore thermocouples for boiling water reactors that augment the reactor vessel water level indicators. The instrumentation system must supply to the control room a recorded, unambiguous, easy-to-interpret, indication of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovery and give advance warning of the approach of inadequate core cooling. (January 1, 1982.)

(D) All instruments used to detect the existence of inadequate core cooling shall be designed and qualified to perform their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section. (January 1, 1982) (II.F.2)

(viii) An analysis shall be provided that defines the potential for voiding in the reactor coolant system during anticipated transients. (Applicable to PWRs only) (January 1, 1982) (II.K.2.17)

(ix) An analysis shall be provided of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater. (Applicable to PWRs only) (January 1, 1982) (II.K.2.19)

(x) If determined necessary as a result of the analysis required by paragraph f(1)(xLi) of this section, an automatic power-operated relief valve isolation system shall be installed that will automatically cause the block valve to close when the reactor coolant system pressure falls after the PORV has opened (Applicable to PWRs only). (This requirement shall be implemented, if found to be necessary, by the end of the first refueling 6 months after staff approval of the design.) (II.K.3.1)

(xi) The automatic depressurization system, valves, accumulators and associated equipment instrumentation shall show to be capable of performing their intended safety functions during and following exposure to the hostile environment of an accident situation, taking no credit for non-safety related equipment or instrumentation, and taking account for air (or nitrogen) leakage through valves. (Applicable to BWRs only.) (January 1, 1982) (II.K.3.28)

(xii) Plant-specific calculations for small break loss of coolant accidents shall be provided consistent with the revised models development pursuant to item f(1)(xLiii) of this section. (January 1, 1983) (II.K.3.31)

(xiii) The RCIC system shall automatically transfer its suction to the suppression pool when the condensate storage tank level is low. (Applicable to BWRs only). (January 1, 1982) (II.K.3.22)

(xiv) The design of the automatic depressurization system shall be such that any operation of this system needed to assure adequate core cooling will be initiated automatically. The design description shall be submitted to the NRC for approval by April 1, 1982, or as part of the FSAR, whichever is later. For operating licenses issued prior to April 1, 1983, the design shall be installed not later than the first refueling outage that is at least six months subsequent to staff approval of the design. For operating licenses issued on or after April 1, 1983, the design shall be installed not later than the start of power operation (i.e., greater than 5% of rated power). (Applicable to BWRs only) (II.K.3.18)

(xv) The core spray and the LPCI systems shall automatically restart upon low water level, if an initiation signal is still present, to assure adequate core cooling. The design description shall be submitted to the NRC for approval. For operating licenses issued prior to January 1, 1982 the design shall be installed not later than the first refueling outage that is at least six months subsequent to staff approval of the design. For operating licenses issued on or after January 1, 1982, the design shall be installed not later than the start of power operations (i.e., greater than 5% rated power). (Applicable to BWRs only) (II.K.3.21)

Dated at Washington, D.C., this 7th day of May 1981.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

Note.—Commissioner Bradford's dissenting views are attached.

10 CFR Part 50

Licensing Requirements for Pending Operating License Applications

Commissioner Bradford dissented from the publication of the proposed rule on the grounds that the subject matter was too broad to be dealt with coherently and effectively in a single rulemaking.

[FR Doc. 81-14392 Filed 5-12-81; 8:45 am]

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SELECTIVE SERVICE SYSTEM

32 CFR Ch. XVI

Improving Government Regulations; Semiannual Agenda

AGENCY: Selective Service System.

ACTION: Semiannual agenda.

SUMMARY: The purpose of this agenda is to report the proposed rulemaking activities of the Selective Service system that might affect the processing of registrants under the Military Selective Service Act (50 U.S.C. App. 451 et seq.). This information will allow the public to participate in the System's decisionmaking at an early stage.

FOR FURTHER INFORMATION CONTACT: Edward A. Frankle, Associate Director, Policy Development Directorate, Selective System, Washington, D.C. 20435. Telephone (202) 724-0844.

SUPPLEMENTARY INFORMATION: This agenda is published in accord with the requirements of E.O. 12291. Selective Service Regulations appear in 32 CFR Chapter XVI.

Subjects of Proposed Rulemaking

Consideration will be given to a comprehensive revision of Selective Service Regulations that deal with the processing of registrants under the Military Selective Service Act (50 U.S.C. App. 451 et seq.). Regulations for the administration by the System of the Freedom of Information Act (5 U.S.C.

PROPOSED O.L. RULE

ON TMI ACTION PLAN (NUREG 0737)

SUMMARY OF SPECIFIC ITEM COMMENTS

Item No. (f)

Additional TMI-related requirements for applications for an operating license. In addition to the requirements of paragraph (b) of this section, each application for an operating license that is to be issued after (* * * insert effective date of this rule * * *) shall meet the requirements in paragraphs (f)(1) and (2) of this section. If the applicant contends that implementation of an item on the schedule set forth in this rule is impractical for its facility, the applicant may provide information to support this contention. The Commission will evaluate this information and, based on its determination of earnest effort and good cause shown, may grant relief from the implementation schedule, on a case-by-case basis. In such cases, the Commission will impose alternative schedule requirements suitable for that facility.

Public Comments

o the word "Commission" should be changed to "NRC staff."

ITEM NO. (1)(i)

(i) The minimum shift staffing for operators, licensed and non-licensed, shall be as shown in Table 1. In addition to the staffing requirements stated in the Table, each operating shift, except during periods of cold shutdown, shall include a qualified Shift Technical Advisor (STA). In addition to the staffing requirements stated above, shift crew assignments shall include a licensed senior reactor operator to directly supervise core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties. The amount of overtime worked by plant staff members performing safety-related functions shall be limited. Other onshift staffing and emergency response capabilities shall be as shown in Table 2. The capability for augmentation of resources for emergency response functions shall be equivalent to that shown in Table 2. (I.A.1.1;[1] I.A.1.3; III.A.1.2)

Table 1.—*Required Shift Manning*

Operating status	One unit, one control room	Two units, one control room	Two units, two control rooms	Three units, two control rooms
One Unit Operating*	1 SS (SRO) 1 SRO 2 RO 2 AO	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 1 SRO 4 RO 4 AO
Two Units Operating*	NA	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 2 SRO 4 RO 4 AO	1 SS (SRO) [1]2 SRO [1]5 RO 5 AO
All Units Operating*	NA	1 SS (SRO) 1 SRO 3 RO 3 AO	1 SS (SRO) 2 SRO 4 RO 4 AO	1 SS (SRO) 2 SRO 5 RO 5 AO
All Units Shut Down	1 SS (SRO) 1 RO 1 AO	1 SS (SRO) 2 RO 3 AO	1 SS (SRO) 2 RO 3 AO	1 SS (SRO) 3 RO 5 AO

SS—shift supervisor.
SRO—licensed senior reactor operator.
RO—licensed reactor operator.
AO—auxiliary operator.
[1] Only 1 SRO and 4 ROs required if both units are operated from one control room.
NOTE —(1) In order to operate or supervise the operation of more than one unit, an operator (SRO or RO) must hold an appropriate, current license for each such unit.
(2) In addition to the staffing requirements indicated in the table, a licensed senior operator will be required to direct supervise any core alteration activity.
(3) See item I.A.1.1 for shift technical advisor requirements.
*Modes 1 through 4 for PWRs.
Modes 1 through 3 for BWRs.

Table 2.—*Minimum Staffing Requirements for NRC Licensees for Nuclear Power Plant Emergencies*

Major functional area and location	Major tasks	Position title or expertise	On shift*	Capability for additions 30 min	Capability for additions 90 min
Plant Operations and Assessment of Operational Aspects.		Shift Supervisor (SRO)	1		
		Shift Foreman (SRO)	1		
		Control Room Operators	2		
Emergency Direction and Control (Emergency Coordinator)***		Auxiliary Operators	2		
		Shift Technical Advisor, Shift Supervisor or designated facility manager.	1**		
Notification/Communication****	Notify licensee, State, local and Federal personnel and maintain communication.		1	1	2
Radiological Accident Assessment and Support of Operational Accident Assessment.	Emergency Operations Facility (EOF) Director	Senior Manager			1
	Offsite Dose Assessment	Senior Health Physics (HP) Expertise			
	Offsite Surveys			1	
	Onsite (out-of-plant)			2	2
	In-plant surveys	HP Technicians		4 1	? 1
	Chemistry/Radio chemistry	Rad/Chem Technicians	1	1	1
Plant System Engineering, Repair and Correcting Actions.	Technical Support	Shift Technical Advisor	1		1
			1		
	Core/Thermal Hydraulics				
	Electrical			1	
	Mechanical				1
	Repair and Corrective Actions				1
		Mechanical Maintenance/Rad Waste Operator	1**		1
		Electrical Maintenance/Instrument and Control (I&C) Technician	1**	1	1
Protective Actions (In-Plant)	Radiation Protection: a. Access Control	HP Technicians		1	
	b. HP Coverage for repair, corrective actions, search and rescue first-aid and firefighting.		2**	2	2
	c. Personnel monitoring				
	d. Dosimetry				
Firefighting					
Rescue Operations and First-Aid			[1]	[2]	[2]
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability.	Security Personnel	2**	[2]	[2]
			[2]		
		Total	10	11	15

[1] Fire Brigade per Technical Specifications.
[2] All per security plan.
[3] Local support.
NOTES:
*For each unaffected nuclear unit in operation, maintain at least one shift foreman, one control room operator and one auxiliary operator except that units sharing a control room may share a shift foreman if all functions are covered.
**May be provided by shift personnel assigned other functions.
***Overall direction of facility response to be assumed by EOF director when all centers are fully manned. Director of minute-to-minute facility operations remains with senior manager in technical support center or control room.
****May be performed by engineering aide to shift supervisor.

Public Comments

- o delete - no flexibility to fit licensee's design or organizational needs (12).
- o delete - emergency planning covered by existing rules and regulatory guides (4).
- o delete - table 1 contained in Technical Specifications (2).
- o delete - STA requirement may be temporary (17)
- o limitation of overtime is too ambiguous (7).
- o time requirements in table 2 will cause problems for plants sited remotely - change to 60 minutes for all (11).
- o do not need technician level Health Physics and Rad Chem personnel - cross train operating personnel (2).
- o requirement for SRO to supervise core alterations should not exclude a foreman with an SRO limited to fuel handling (2)
- o delete - STA training requirement details have not been established.
- o STA not required during cold shutdown, but that may be when he is needed most due to abnormal lineups/tests.
- o previous guidance was that STA did not have to be actually on shift, but within 10 minute call.

Item No. (1)(ii)

The operator initial training and requalification programs shall include: heat transfer, fluid flow, and thermodynamics; and emphasis on reactor and plant transients.

The training program for all operating personnel shall include training to recognize, control and mitigate the consequences of accidents in which the core is severely damaged. In addition, each applicant shall support the development of its training program, emergency procedures and control room hardware, with applicable human engineering data. Additionally, intensive and comprehensive training exercises are to be conducted during low-power testing programs to provide experience for each operating shift. The principal instructors shall be qualified at the senior reactor operator level and shall periodically thereafter demonstrate their continued competency. An applicant for a senior reactor operator license shall have had experience as an operator and shall participate in an NRC approved training program. In addition to the written examination and the oral examination administered in the plant, and (SIC) operational examinations on an appropriate simulator will be administered by the NRC. The minimum passing grade shall be 80% overall with a minimum in each technical category of 70%. (I.A.2.1; I.A.2.3; I.A.3.1; I.G.1; II.B.4)

Public Comment

o consideration of the importance of the tested area should dictate the minimum grades required.

o human engineering data is relatively non-existent - term requires definition (6).

o delete - can be covered in 10CFR55 and Appendix A (4).

o add provision for use of site specific simulators for training (4).

o add that minimum grades are for written examinations.

o delete - NRC does not have resources to administer simulator examinations (3).

o delete - requirement for instructors to be SRO's (5).

o change requirement for operator training to apply to SRO's and RO's only - not "all operating personnel" (8).

o operational exams on non-plant specific simulators can have a negative effect on actual performance.

o no provision to ensure honesty and accuracy of administered examinations.

o delete requirement that SRO be RO first. This postpones the entry of college graduates and other capable individuals. Because of unions, college graduate would have to start at bottom - delaying move to management - this discourages them from field of operations.

o it is not clear that the 80% overall score is justified in light of the 70% required in each area.

Item No. (1)(iii)

Corporate management directives shall be issued that emphasize the shift supervisor's role in the control room as the primary onsite manager responsible for safe operation of the plant under all conditions. Such directives shall clearly define his responsibilities and authority including his command decision authority, relative to other plant management personnel, over plant operations personnel. The shift supervisor's responsibilities shall include limiting personnel access to the control room during emergencies; his administration duties shall be such that they do not detract from or are subordinate to the management responsibility for assuring the safety operation of the plant. Training programs for shift supervisors shall strengthen both management and operational capabilities. (I.A.1.2; I.C.3; I.C.4)

Public Comment

- o it is inappropriate to mandate the Shift Supervisor as being responsible for access to the control room during emergencies - use administrative procedures (7).
- o delete - requirement is incorporated in ANS 3.2 and endorsed by Reg. Guide 1.33 (2).
- o shift supervisor does not need to be in "the control room" to perform duties as "primary on-site manager."

Item No. (1)(iv)

An onsite independent safety engineering group of technically qualified personnel shall be provided to perform continuing systematic reviews of plant activities, including operating experience information that may indicate areas for improving plant safety. This group shall also provide recommendations and advice to an offsite high level corporate technical officer, not in the management chain for power production. (I.B.1.2)

Public Comments

- delete - ISEG is interim - will be reviewed to determine its effectiveness in $\sim$ 1 year (8).
- delete - functions of the ISEG already being performed by various groups in the organization (3).
- delete provisions to report to off-site personnel dilutes responsibility and authority of those ensuring safety.
- delete - put in Technical Specifications and Reg. Guide 1.33 (2).
- delete requirement for offsite officer not to be in the power production chain of command.
- allow members of ISEG to be assigned to offsite locations (2).
- no guidance on what constitutes technically qualified personnel (2).
- not comprehensive enough (2).
- delete the requirement that the high level officer be "offsite" (2).

Staff Comments

- no comments.

Item No. (1)(v)

Analyses of small-break-loss-of-coolant accidents and of transients and accidents that involve postulated multiple failures, consequential failures, and operator errors, which if unmitigated could lead to inadequate core cooling, shall be provided. The analyses shall be carried sufficiently into the event to identify all significant thermal/hydraulic/neutronic phenomena and to address possible failures and operator errors during the long-term cooling phase. Emergency procedure guidelines to mitigate these transients and accidents shall be provided. (I.C.1)

Public Comment

- o delete - too general - intervenors could extend licensing hearings ad infinitum (8).
- o G.E.'s analysis was a one-time effort and is complete.
- o delete - already covered in 10CFR50.36.

Item No. (1)(vi)

Administrative controls shall be provided to ensure adequate exchange of plant status information between control room operations personnel during shift and relief turnover. As a minimum, the exchanged information shall include: Values of key plant parameters, availability and alignment of systems important to safety, identification of systems and components in an acceptable degraded mode of operation, and identification of systems out of service for maintenance or test. (I.C.2)

Public Comments

- o delete - already covered in Reg. Guide 1.33.
- o delete second sentence as first sentence adequate covers intent of NUREG-0660.

Item No. (1)(vii)

A management system shall be provided to perform the following functions: (A) Review operating experience information originating both within and outside the facility; (B) Promptly supply information pertinent to plant safety, including proposed procedural changes and plant modifications, to operators and other appropriate plant personnel; and (C) Assure that such information is incorporated into training and requalification programs. (I.C.5)

Public Comments

- o delete the word proposed in (B) - inform upon implementation.
- o delete - these requirements are included elsewhere (?).

Item No. (1)(viii)

A management system shall be provided to independently verify the proper performance of operational and maintenance activities, as a means of reducing errors that could result in or contribute to accidents. The system shall include automatic status monitoring or verification by two qualified individuals. (I.C.6)

Public Comments

- delete - inconsistent with considerations of reducing exposure to individuals (7).
- delete last sentence - too prescriptive (2).
- delete - too broad.
- delete - not required in NUREG-0737.
- delete - the requirement that "qualified personnel" perform independent verifications on redundant systems.
- delete - management system far too confining (3).

Item No. (1)(ix)

Reviews of the proposed procedures for low-power test, power ascension tests, and emergency procedures to verify the adequacy of procedures shall be obtained from the nuclear steam system supplier. (I.C.7)

Public Comments

- o delete - large amount of money would have to be expended to obtain verification from NSSS vendor.
- o make it clear that NSSS vendor provides comments, but licensee is responsible for evaluation and determination of adequacy.
- o NSSS review may not be documented.
- o NSSS vendor does not have all details of each plant and is not informed of revisions.
- o power accension test procedures can be verified prior to test performance - there should not be a requirement to perform review prior to license issuance.
- o delete - rules should not be used to require reviews (2).

Item No. (1)(x)

Detailed reviews of the final design shall be performed to ensure that the design of the control room and control boards are in conformance with good human factors engineering principles and that information for the control room operators is presented in a manner that facilitates recognition of developing off-normal conditions, and mitigation of accidents. (I.D.1)

Public Comments

- no acceptance criteria exists for compliance with this rule (4).
- delete - rules should not be used to require reviews.
- not specific enough.

Item No. (1)(xi)

A plan for collection of data shall be provided that will establish for ECCS systems and equipment: (A) Outage dates and durations, (B) Cause of the outage, (C) Systems or components involved, (D) Specific corrective actions taken, and (E) Changes that may improve ECCS equipment availability. (II.K.3.17)

Public Comments

- delete - requirement already contained in Technical Specifications (5).
- delete - the collection of this data is a one time effort - does not need to be codified (3).
- change "plan" to "plan and/or procedures."
- too limited - should include all engineered safety features, reactor scram and associated instruments, and auxiliary supporting systems.

Item No. (1)(xii)

Direct position indications (open or closed) for the relief and safety valves shall be provided in the control room. (II.D.3)

Public Comments

- o Replace "Direct" with "Positive" (8).
- o Need to define "Direct position indication."
- o Not clear that this applies to pressurizer valves only - not all relief valves in the plant (6).
- o delete - already included in standard Technical Specifications.

Item No. (1)(xiii)

The auxiliary feedwater system (AFW) shall be evaluated including: (A) A simplified AFW reliability analysis using event-free and fault-free logic techniques; (B) A design review of AFW; and (C) An evaluation of AFW flow design bases and criteria. (Applicable to PWRs only.)(II.E.1.1)

Public Comments

- o delete - design review and evaluation is usual for a safety grade system such as AFW.
- o delete - rules should not be used to require reviews.
- o delete - already covered in construction permit requirements.
- o should apply to all decay heat removal systems and acceptance criteria should be established.

Item No. (1)(xiv)

The protection system shall include automatic and manual initiation of the auxiliary feedwater system and control room indication of system flow. (Applicable to PWRs only.)(II.E.1.2)

Public Comments

- o delete - detailed requirements inconsistent with other Commission regulations (2).
- o delete - already covered in elsewhere (6).

Item No. (1)(xv)

The design shall include the capability to promptly connect onsite electric power to: (A) Pressurizer heater and associated controls sufficient to establish and maintain natural circulation in hot standby conditions, (B) Pressurizer power-operated relief valves, (C) The block valves for the pressurizer power-operated relief valves, and (D) pressurizer water level instrumentation. (Applicable to PWRs only.)(II.E.3.1 & II.G.1)

Public Comments

- delete - to prescriptive - would require all PORV's to be Class 1E - B&W designs upstream block valve to be 1E (2).
- should use "emergency buses" vice "onsite electric power".
- delete - should be covered in Regulatory Guide or Standard Review Plan.
- does not provide additional protection unless these components are classified as "important to safety."

Item No. (1)(xvi)

Each power reactor that relies upon external recombiners or purge repressurization systems to satisfy the requirements of §50.44 of this part shall be provided with containment penetrations for the external recombiners or purge/repressurization systems that either: (A) Are dedicated to that service only, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures and are sized to satisfy the flow requirements of the external recombiners or purge/repressurization systems, or (B) Are of a combined design for use by either external recombiners or purge/repressurization systems and other sys ms, conform to the requirements of Criteria 54 and 56 of Appendix A of this part, are designed against postulated single failures both for containment isolation purposes and for operation of the external recombiners or purge/pressurization systems, and are sized to satisfy the flow requirements of the external recombiners or purge/pressurization systems. (II.E.4.1)

Public Comments

o delete - hydrogen criteria given in 10CFR50.44 and Appendix A - this should be there (9).

Item No. (1)(xvii)

The containment isolation system design shall provide that: (A) All non-essential systems are isolated automatically, (B) Each non-essential penetration (except instrument lines), has two isolation barriers in series, (C) The overriding (resetting) of the isolation signal shall require deliberate operator actions of at least two steps and no single sequence of operator override actions shall cause the reopening of the containment penetrations associated with more than one system or more than one purge or vent isolation valve, (D) The containment high pressure set point for initiating containment isolation is as low as is compatible with normal operation, and (E) All containment purge and vent isolation valves will receive an automatic closure signal on containment high radiation. (II.E.4.2)

Public Comments

- o Item A - should recognize that certain exceptions are allowed.
- o Item C - delete "overriding (resetting)" substitute "resetting."
- o Item E - delete "containment high radiation" substitute "high radiation."
- o delete - covered in 10CFR50, Appendix A (8).
- o Item E has not been determined to be required - should be changed to - "an evaluation shall be provided to determine the feasibility and effectiveness of automatically closing the containment vent and purge isolation valves on a containment high radiation signal (4).
- o some "non-essential" systems e.g., main feedwater may be of value in responding to transients - should be allowed to remain available.
- o Item D - provides no specific requirement.

Item No. (1)(xviii)

A review shall be provided of all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper Engineered Safety Features (ESF) functioning. (II.K.1.5)

Public Comments

- delete - covered elsewhere/should be covered elsewhere (2).
- delete - thrust of requirement is unclear.
- delete - rules should not require reviews.

Item No. (1)(xix)

Procedures for removing safety-related systems from service (and restoring to service) shall be provided that ensure that operability status will always be known by the control room operators. (II.K.1.10)

Public Comments

- o delete - already covered elsewhere (4).
- o vague and inadequate.
- o safety related systems should not be removed from service unless plant conditions prevent an accident requiring the system from occurring.

Item No. (1)(xx)

Safety injection shall be initiated when the pressurizer low pressure setpoint is reached regardless of the pressurizer level. (Applicable to Westinghouse-designed reactors only.) (II.K.1.17)

Public Comments

- o delete - hardware fix without defining problem on criteria - already has been accomplished.
- o should apply to all PWRs.
- o should state that pressurizer level indication shall not be used to inhibit automatic initiation of safety systems.

Item No. (i)(xxi)

The reactor protection system shall include anticipatory reactor trip for loss of main feedwater, turbine trip, and significant decrease in steam generator level. Procedures and associated operator training shall be provided to ensure prompt manual reactor trip for main steamline isolation valve closure, loss of offsite power, and low pressurizer level. (Applicable to B&W-designed reactors only.) (II.K.1.20, II.K.1.21, and II.K.2.10.)

Public Comments

o applicants should be permitted to identify anticipatory trips which are the most appropriate for their own facilities (4).

o reconsider - requires plant modification.

o remove "anticipatory" - it provides a basis for evading the existing regulations applicable to protection systems.

o manual trips should be automatic - cannot rely on the operator.

Item No. (1)(xxii)

An analysis shall be provided to verify that the power-operated relief valves on the pressurizer will open during less than five percent of all anticipated over-pressure transients for the range of plant conditions which might occur during a fuel cycle. (Applicable to B&W-designed reactors only.)(II.K.2.14 and II.K.3.7)

Public Comments

o too specific - should allow other way of providing safety to PORV's (2).

o delete - rules should not require analysis.

o PORV should be classified as safety grade.

Item No. (1)(xxiii)

The design of the auxiliary heat removal systems shall be such that necessary automatic actions will occur, and manual actions can be taken, when the main feedwater system is not operable. (Applicable to BWRs only.)(II.K.1.22)

Public Comments

o should be in 10CFR50, Appendix A (6)

o inconsistent with intent of requirement II.K.1.22 of NUREG-0660 (5).

o should be classified as system important to safety and instruments part of Reactor Protection System.

Item No. (1)(xxiv)

A description shall be provided of all reactor vessel level indications used for automatic or manual initiation of safety systems. Other instrumentation that might give the operator the same information on plant status shall also be described. (Applicable to BWRs only.)(II.K.1.23)

Public Comments

- o was a generic - one time effort - any requirements should be included in appropriate regulations (6).
- o inadequate - system should be in RPS and should apply to PWR's also.

Item No. (1)(xxv)

Procedures and training shall be provided for operating personnel relative to initiation and control of auxiliary feedwater independent of the Integrated Control System. (Applicable to B&W-designed reactors only.)(II.K.2.2)

Public Comments

- delete - ICS plays no role in AFW initiation and control for new B&W designs (2).
- delete - too detailed/limited applicability.
- inadequate and vague.

Item No. (1)(xxvi)

A failure modes and effects analysis of the integrated control system (ICS) shall be provided. (Applicable to B&W-design reactors only.)(II.K.2.9)

Public Comments

- o why is rule governing control system FMEA limited to one reactor supplier?
- o too detailed/limited applicability (2).
- o delete - rules should not require analysis and evaluations.
- o no requirement for the submittal of information similar to that already supplied should be made until NRC has reviewed the information already provided.

Item No. (1)(xxvii)

A detailed analysis of thermal-hydraulics conditions in the reactor vessel during recovery from a small-break LOCA, with an extended loss of all feedwater, requiring the use of the cooler high-pressure injection system water, shall be provided to confirm that vessel integrity is not jeopardized. (Applicable to PWRs only.) (II.K.2.13)

Public Comments

- o soften - differences in vessel material and design may not warrant detailed analysis.
- o delete - rules should not require analysis and evaluations.
- o delete - should be addressed in subsidiary Commission guidance pertaining to the implementation of GDC-31.
- o not required in all cases - due to vessel material and design differences.

Item No. (1)(xxviii)

An analysis shall be provided of the effects of slug flow on the once-through steam generator tubes after primary system voiding. (Applicable to B&W-design plants only.) (II.K.2.15)

Public Comment

- o delete - rules should not require analysis and evaluation
- o no requirement for the submittal of information similar to that already supplied should be made until NRC has reviewed the information already provided (2).

Item No. (1)(xxix)

An evaluation shall be provided of the potential for and impact of reactor coolant pump seal damage and leakage upon loss of offsite power. If such damage is indicated, an analysis shall be provided of the limiting small-break loss-of-coolant accident complicated by subsequent reactor coolant pump seal damage. (II.K.2.6 and II.K.3.25)

Public Comments

- o delete - can be covered in 10CFR50 Appendix A or elsewhere (2).
- o should not be required if leakage through the seal is less than makeup capacity.
- o delete - rules should not require analysis and evaluations.

Item No. (1)(xxx)

For Westinghouse-designed facilities where the reactor trip is to be bypassed when operating below 50 percent power, an evaluation shall be provided to verify that the probability of a small break LOCA resulting from a stuck-open PORV is not significantly greater than the case where this trip is bypassed only when operating below 10 percent power. (II.K.3.10)

Public Comments

- o not clear if trip in question is the anticipatory trip on the turbine trip (2).
- o delete - Westinghouse studies show the probability of challenging the PORV's was not significantly increased due to the bypass of reactor trip on turbine trip below 50% power.
- o too detailed/limited applicability (2).
- o delete - rules should not require analysis and evaluation.

Item No. (1)(xxxi)

An analysis shall be provided that defines the probability of a small-break LOCA caused by a stuck-open power operated relief valve (PORV). If this probability is a significant contributor to small-break LOCAs from all causes, provide a design description for an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWRs only.)(II.K.3.2 and II.K.3.1)

Public Comments

- o delete reference to Item II.K.3.1 in NUREG-0737.
- o implementation covered under 10CFR50.34(f)(2)(x).
- o delete-rules should not require analysis and evaluation.
- o future guidance should be issued in form of subsidiary guidance to GDC-14.

Item No. (1)(xxxii)

Any failure of a safety or relief valve shall be reported promptly to the NRC and all challenges to such valves shall be reported annually. (Applicable to PWRs only.)(II.K.3.3)

Public Comments

- o reports should be via LER system (2).
- o delete - not appropriate and has no precedence as a rule (5).
- o Reporting safety and relief valve failures is already required by 10CFR21 (6).
- c should indicate if applicable to all LWR's as in NUREG-0737.
- o not clear which safety and relief valves.

Item No. (1)(xxxiii)

An evaluation shall be provided of the automatic tripping of the reactor coolant pumps in the case of a small-break loss-of-coolant accident. (Applicable to PWRs only.)(II.K.3.5)

Public Comments

- o delete - implementation has been deferred by NRC pending completion of LOFT test L3-6 blind post-test analyses (2).
- o delete - temporary solution until a better one is found (2).
- o delete - rules should not require analysis and evaluations.
- o applicant should be allowed to show the RCP trip is not necessary or that if necessary sufficient time is available to allow for manual tripping (2).

Item No. (1)(xxxiv)

If a proportional integeral-derivative controller is installed in the power operated relief valve (PORV) control system, the control system shall be operated so as to preclude opening the PORV due to derivate action. (Applicable to Westinghouse-designed reactors only.)(II.K.3.9)

Public Comments

o delete - hardware fix without defining problem or criteria - already has been accomplished.

o delete - too detailed/limited applicability.

Item No. (1) (xxxv)

(xxxv) Complete justification shall be provided for the use of any type of pressure-operated relief valve that has failed during testing (such as those supplied by Control Components, Inc., that failed during hot functional testing at a plant). (Applicable to PWRs only.) (II.k.3.11)

Public Comments

- o Example cited should be deleted as the intent is clear without it
- o delete - should be addressed as part of QA program rule
- o delete - to detailed/limited applicability (3)
- o Vague and inadequate

Item No. (1) (xxxvi)

(xxxvi) An anticipatory reactor-trip on turbine-trip shall be provided. (Applicable to Westinghouse-designed reactors only.) (II.k.3.12)

Public Comments

- o Should have provision for turbine-trip without reactor trip at low power (10-15%) to reduce the number of reactor trip cycles
- o delete - some plants have 100% load rejection capability and don't need a reactor trip following a turbine trip to maintain safety requirements (NOTE: will result in more frequent challenges to PORV's)

Item No. (1) (xxxvii)

(xxxvii) An evaluation shall be provided of the safety effectiveness of initiating the reactor core isolation cooling system at a higher water level than that for the high pressure coolant injection system and of restarting both systems on low water level. (Applicant (SIC) to BWRs only.) (II.k.3.13)

Public Comments

- o should be modified to reflect original intent of NUREG-0737 (reducing the number of thermal cycles to reactor vessel and intervals resulting from HPCI initiating) (4)
- o delete - rules should not require evaluations
- o delete - generic information submitted by BWR Owner's Group to NRC for review

Item No. (1) (xxxviii)

(xxxviii) The design of the HPCI/RCIC steam line pipe-break-detection circuitry shall be such that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent isolation of these systems. (Applicable to BWRs only.) (II.k.3.15)

Public Comments

- o needs clarification - applicable to BWR 6 with HPCS vice HPCI?
- o delete - generic data submitted by BWR Owners Group for NRC review (2)
- o delete - could be addressed in 10 CFR 50, Appendix A

Item No. (1) (xxxix)

(xxxix) An analysis shall be provided to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems, shall be provided. (Applicable to BWRs only.) (II.k.3.16)

Public Comments

- o delete - generic date supplied by BWR Owner's Group for NRC review (3)
- o delete - last three words as they are superficious
- o delete - Rules should not require analysis and evaluations

Item No. (1) (xl)

(xl) Pending the implementation of automatic transfer features, clear and cogent procedures shall exist for manual transfer of RCIC system suction to the suppression pool when the condensate storage tank level is low. (Applicable to BWRs only.) (II.k.3.22)

Public Comments

- o delete - not appropriate to require interim operating procedures in a rule (4)
- o delete - RCIC transfer procedures should be located with the item on automatic RCIC transfer [(f)(2)(xiii)] (2)

Item No. (1) (xli)

(xli) The HPCI and RCIC systems shall be designed to withstand and operate satisfactorily following a complete loss of offsite power for at least two hours. (Applicable to BWRs only.) (II.K.3.24)

Public Comments

- o delete - covered in 10 CFR 50, Appendix A.
- o modify to reflect original intent of NUREG-0737 (maintain pump room temperature limits within limits on loss of offsite power).
- o clarify applicability to BWR 6 with HPCS vice HPCI system.

Item No. (1) (xlii)

(xlii) The scales of the [illegible] reactor vessel water level instruments shall be referenced to the [illegible] int. (Applicable to BWRs only.) (II.K.3.27)

Public Comments

- o delete - covered in 10 CFR 50, Appendix A (6)
- o Add scope of item II.K.3.27 to criterion 13 in 10 CFR 50, Appendix A. (5)
- o delete the words "The scales of" and "various" (4).
- o Some indicators perform special functions which require referencing to a different point.

Item No. (1) (xliii)

(xliii) Small-break loss-of-coolant accident analysis methods used to comply with Appendix K to 10 CFR Part 50 shall be revised and provided that account for experimental data, including data from the Loss-of-fluid-test (LOFT) and Semiscale Test facilities. This evaluation shall consider the LOFT test, (L3-6). (II.K.3.30)

Public Comments

- delete - could be covered in 10 CFR 50, Appendix A (4).
- delete last sentence and change first sentence to read "... account for applicable experimental data." (6)
- justification of current models should be permitted (4).
- delete - already covered in 10 CFR 50, Appendix K (6).
- delete - was a generic, one-time effort. Section 2 item xii provides adequate followup.
- delete until NRC has complete review of B&W models.
- delete - rules should not require analysis and evaluations.
- delete - has been completed for PWR NSSS and little or no further data is required.

Item No. (1) (xliv)

(xliv) Analysis shall be provided to demonstrate that for anticipated transients complicated by the worst single failure, and assuming proper operator actions, the core remains covered or no significant fuel damage results from core uncovery. (Applicable to BWRs only). (II.K.3.44)

Public Comments

- o does not analyze for multiple failure (i.e. mechanical and human).
- o delete - requiring analysis for transient plus a single failure goes beyond current design basis as given in 10 CFR 50, Appendix A, Criteria 10 and 20 (4).
- o delete - "Applicable to BWRs only" (4).
- o delete - could be specified in Standard Review Plan, Section 15.6.
- o delete - rules should not require analysis and evaluations.
- o delete - generic data supplied by BWR Owners Group for NRC review.

Item No. (1) (xlv)

(xlv) Analysis shall be provided to support depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWRs only.) (II.K.3.45)

Public Comments

- o reword - "Provide analysis to examine the reduction in vessel stresses and the impact on core cooling margin that results due to slower depressurization rates for the automatic depressurization system." (5)
- o BWR Owners Group submitted the position that alternate depressurization methods would not enhance safety of the plant. (Approved by NRC) (7).
- o Vague and inadequate.

Item No. (1) (xlvi)

(xlvi) Each boiling and pressurized light-water nuclear power reactor applicant shall implement leak reduction measures to that leakage, from systems outside containment (systems that would or could contain highly radioactive fluids during and following a serious transient or accident), is eliminated or minimized to the maximum extent practicable to prevent the release of significant amounts of radioactive material during and following an accident. Consideration shall be given to reductions of potential release paths that could result from design or operator deficiencies.

Public Comments

- o delete - can be covered in 10 CFR 50, Appendix A (7).
- o wording not clear - changing to wording and intent of NUREG-0737 Item III.D.1.1 preferred (7).
- o delete last sentence of paragraph.
- o should read "... measures so that leakage ...".

Item No. (1) (xlvii)

(xlvii) Each boiling and pressurized light-water nuclear power reactor licensee shall establish and implement a program of preventive maintenance to eliminate or minimize, to the maximum extent practicable, leakage from systems outside containment. This program shall include periodic (integrated) leak tests of these systems at intervals not to exceed each refueling cycle and also include (as-well-as) the reduction of potential release paths by appropriate operator training. (III.D.1.1)

Public Comments

- o delete - intent of first sentence in embodied in proposed rule (xlvi) (7).
- o delete the word "integrated".
- o delete - can be covered elsewhere.
- o vague and inadequate.

Item No. (1) (xlviii)

(xlviii) Each boiling and pressurized light-water power reactor shall be provided with instrumentation, equipment and associated training and procedures for determining, under accident conditions, the airborne radioiodine concentration in areas within the facility where plant personnel may be present during and following an accident. (III.D.3.3)

Public Comment

- No comment

Item No. (1) (xlix)

(xlix) The control room and associated habitability systems shall be designed to adequately protect the reactor operations staff against the effects of accidental release of toxic or radioactive gases such that the nuclear plant can be operated or safety (SIC) shutdown under accident conditions. Analysis based upon the final as-built conditions shall be provided to demonstrate that airborne concentrations of such hazardous fumes will permit control room operators to remain in the control room to take appropriate safety actions. (III.D.3.4)

Public Comments

- o delete - already covered in 10 CFR 50, Appendix A (11).
- o delete - NUREG-0737 list documents which establish NRC requirements on this concern (all pre-TMI) and admits there is no change.
- o vague and inadequate.

Item No. (1) (1)

(1) Dedicated emergency response facilities shall be established and maintained for command and control, support, and coordination of onsite and offsite functions during reactor accident conditions. The Technical Support Center is to provide an appropriate near-the-control-room location for those individuals who are knowledgable of and responsible for engineering and management support of reactor operations, to diagnose and evaluate plant conditions and for more orderly conduct of plant activities during emergency conditions. The Operational Support Center is to provide an area separate from the control room for shift and other support personnel (e.g., auxiliary operator, technicians, health physics personnel) to report for instructions from the control room staff. The near-site Emergency Operations facility is to provide (A) a center for analysis of plant effluents, meteorological conditions, offsite radiation measurements and for offsite dose projections, and (B) a center for coordination of all licensee onsite and offsite activities and coordination with Federal, State, and local authorities for implementation of offsite emergency plans.

Public Comments

- o does not distinguish between interim capability and (6) upgraded capability (presently required by October 1, 1982).
- o delete - incorporate in 10 CFR 50, Appendix E (3).
- o define "dedicated" - one facility per unit, per site or one per utility?

Item No. (1) (1i)

(1i) Plans and facilities for coping with emergencies and schedules shall be in accordance with the requirements set forth in other sections of 10 CFR Part 50. (III.A.1.1; III.A.1.2; III.A.2)

Public Comments

- o delete - does no more than reference existing requirements already codified (10).
- o delete - too general - does not identify portions of 10 CFR 50 to be complied with.

Item No. (1) (lii)

(lii) The design shall ensure the capability of natural circulation in the event that depressurization of the reactor vessel, during a small break LOCA, is required (II.K.3.46).

Public Comments

o add - applicable to BWRs only (as in NUREG-0737) (3).

Item No. (1) (liii)

(liii) A feasibility study and risk assessment shall be submitted that defines the optimal approach for eliminating the need for manual actuation of the automatic depressurization system to assure adequate core cooling. (Applicable to BWRs only) (II.K.3.18)

Public Comments

- o delete - analysis has been submitted by BWR Owner's Group to NRC.
- o should reference II.K.3.18.a only.
- o delete - rules should not be used to require studies.

Item (2) (i)

(i) Emergency procedures shall be provided to mitigate small-break loss-of-coolant accidents, and transients and accidents that involve postulated multiple failures, consequential failures, and an (SIC) operator errors, which, if unmitigated, could lead to inadequate core cooling. (January 1, 1982) (I.C.1)

Public Comments

- o delete - too broad in scope - could require procedure for every conceivable accident or operator error (2).
- o delete "small-break loss-of-coolant accidents" (6).
- o Criteria for multiple failures to be postulated are not specified (3).
- o Should reference Items I.C.1.2.b and I.C.1.3.b in NUREG-0737 not I.C.1 in toto.
- o To be consistent with NUREG-0737-implementation should be first refueling outage after January 1, 1982 (2).
- o delete - short term upgrading - covered in 10 CFR 50.36.
- o Transient and accident conditions analyzed go beyond the design basis of the plant.

Item No. (2) (ii)

(ii) Each boiling and pressurized light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system and reactor vessel head and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause their loss of function, remotely operated from the control room, to provide improved operational capability to maintain adequate core cooling following an accident. High point vents are not required, however, for the tubes in U-tube steam generators. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents and associated controls, instruments and power sources must conform to the requirements of Appendix A and Appendix B to this Part 50. In particular, the vent system shall be designed to ensure a low probability that (a) the vents will not perform their safety functions and (b) there would be inadvertent or irreversible actuation of a vent. Furthermore, the use of these vents during and following an accident must not aggravate the challenge to the containment or the course of the accident. (July 1, 1982) (II.B.1)

Public Comments

- delete the last two sentences - design details inappropriate for regulation.

Item No. (2) (iii)

(iii) Each boiling and pressurized light-water nuclear power reactor shall be provided with both adequate access to areas that may be used during and following an accident and protection of safety equipment so that an accident that results in the release of large amounts of radioactive material will not limit personnel occupancy or degrade safety equipment by the radioaction (SIC) fields that may exist during and following the accident to the extent that required safety functions cannot be accomplished.

(A) The facility design must be based on a release of radioactive material from the fuel to the primary coolant system that is not less than 100% of the core equilibrium noble gas inventory, 50% of the core equilibrium halogen inventory, and 1% of the remaining core fission products. For equipment and areas affected by the reactor coolant, it shall be assumed that the above distribution of radioactive material is intimately mixed with the coolant water except that recirculated, depressurized coolant water may be assumed to contain no noble gases. For equipment and areas affected by the containment atmosphere, it shall be assumed that not less than 100% of the core equilibrium noble gas inventory and 25% of the core equilibrium halogen inventory are uniformly dispersed in the containment atmosphere and an additional 25% of the core equilibrium halogen inventory and 1% of the remaining core fission products are uniformly distributed on surfaces exposed to the containment atmosphere.

(B) The facility design basis must be such that an individual operator will not receive more than a 5 rem whole body dose, or its equivalent to any part of the body, while performing a necessary safety function during and following an accident. (January 1, 1982) (II.B.2)

Public Comments

- o delete - percentages proposed have not stood the test of time and scientific examination - use present Regulatory Guide.
- o Analyze realistic scenarios and arrive at envelope values for equipment and compartments.
- o delete - conflicts with 10 CFR Part 20.
- o delete - requirements being resolved in the degraded core rulemaking.

Item No. (2) (iv)

(iv) Each boiling and pressurized light-water nuclear power reactor shall be provided with the capability for personnel to obtain and quantitatively analyze a reactor coolant or containment atmosphere sample during and following an accident.

(A) The facility design must be based on the radioactive material release terms described in paragraph (f) (2)(iii) of this section.

(B) The design basis for the plant equipment that provides the capability to obtain and analyze a sample must be based on the assumption that it will be done promptly, and without incurring a radiation exposure to any individual in excess of 5 rem to the whole body, or its equivalent to any part of the body.

(C) The capability to quantitatively analyze a sample must be based on the use of either in-line monitoring or an onsite radiological and chemical analysis facility. If in-line monitoring is chosen, a capability must be provided for backup sampling using grab samples, and must include the capability for analyzing the samples at either an onsite or offsite facility. The analysis capability must provide, as needed, quantification of the following:

(1) Those radioisotopes necessary to indicate the extent of core damage;

(2) Hydrogen in the containment atmosphere;

(3) Total dissolved gases or dissolved hydrogen gas in the reactor coolant;

(4) Boron in the reactor coolant; and

(5) Chloride in the reactor coolant.

Chloride analyses may be performed offsite and are not required to be done promptly. (January 1, 1982) (II.B.3)

Public Comments

- o delete requirement for sampling chloride in the reactor coolant system for plants using only fresh water heat sinks and makeup. (2)
- o Change (iv)(A) to read (f)(2)(iii) via (f)(1)(iii) (3).
- o delete requirement to sample boron in BWRs.
- o delete requirement to sample for radioisotopes - they determine extent of clad damage NOT core damage.

Item No. (2) (v)

(v) Qualification tests shall be conducted on the reactor coolant system relief and safety valves and, for PWRs, block valves, for all fluid conditions under operating conditions, transients and accidents. Block valves for each relief valve shall be qualified to isolate not only a leaking relief valve under normal conditions, but also any fluid flow conditions generated by a stuck-open relief valve under normal operating or accident conditions. The results of the qualification tests shall be submitted. (Applicable to PWRs only) (July 1, 1982) (II.D.1)

Public Comments

- Reword to be consistent with NUREG-0737 Item II.D.1. (6)
- should be applicable to BWRs also (6).
- delete - adequate data exists concerning these valves.
- delete - GDC-14 already requires testing of reactor coolant pressure boundary - any further requirements should be incorporated in GDC-14.

Item No. (2) (vi)

(vi) Accident Monitoring Instrumentation shall be provided for each boiling and pressurized light-water nuclear power reactor and shall have the capability during and following an accident for:

(A) Providing and recording in the control room a continuous indication of:

(1) Containment pressure;

(2) Hydrogen concentration in the containment atmosphere;

(3) Containment water level;

(4) Containment radiation level; and

(5) Radioactive noble gas concentrations in the plant gaseous effluents at all potential accident release paths effective.

(B) Quantifying the concentration of radioiodines and radioactive particulates in plant gaseous effluents at all potential accident release paths.

(c) Performing their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section. (January 1, 1982) (II.F.1)

Public Comments

- o Does not accurately reflect intent of NUREG-0737 Item II.F.1 (2).
- o Revision to General Design Criteria 64 would be more practical alternative.
- o Record data at location other than control room.
- o delete - too confining.
- o delete - covered in proposed rule or degraded core rulemaking.
- o Modify to defer strict compliance until reasonable time after "qualified" instrumentation is available (2).

Item No. (2) (vii)

(vii) (A) Each boiling and pressurized light-water nuclear power reactor licensee shall develop and implement procedures and training to be used by the operators to recognize the existence of inadequate core cooling and low coolant level in the reactor core using available instrumentation.

(B) Each pressurized light-water nuclear power reactor shall be provided with a primary coolant saturation meter (subcooling meter) that provides in the control room a continuous, recorded, on-line indication of the primary coolant saturation condition.

(C) Each boiling and pressurized light-water nuclear power reactor shall be provided with an instrumentation system, for example, reactor vessel water level indicators for pressurized water reactors that augment the incore thermocouples; and incore thermocouples for boiling water reactors that augment the reactor vessel water level indicators. The instrumentation system must supply to the control room a recorded, unambiguous, easy-to-interpret, indication of inadequate core cooling. The indication must cover the complete range from normal operation to complete core uncovery and give advance warning of the approach of inadequate core cooling. (January 1, 1982)

(D) All instruments used to detect the existence of inadequate core cooling shall be designed and qualified to perform their function following an accident characterized by the radioactive material release terms described in paragraph (f)(2)(iii) of this section. (January 1, 1982) (II.F.2)

Public Comments

- o Applicant should be permitted to determine best means of monitoring inadequate core cooling (2).
- o Instrument should be prominently displayed and able to perform under ext·eme heat conditions.
- o Does not represent position of NUREG-0737 Item II.F.2 (10).
- o Technical justification for requiring incore thermocouples for BWRs has not been provided (8).
- o Delete reference to core coolant level - feel indication can not be either useful or provides unambiguously (2).
- o Support (A) - feel (B), (C) and (D) covered in Regulatory Guide 1.97, Revision 2.

Item No. (2) (viii)

(viii) An analysis shall be provided that defines the potential for voiding in the reactor coolant system during anticipated transients. (Applicable to PWRs only) (January 1, 1982) (II.K.2.17)

Public Comments

- o delete - no additional information needed.
- o delete - rules should not require analysis.
- o delete - specifically tailored to relieve short term consideration.

Item No. (2) (ix)

(ix) An analysis shall be provided of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater. (Applicable to PWRs only) (January 1, 1982) (II.K.2.19)

Public Comments

- delete - irrelevant to B&W plants.
- delete - not applicable to Westinghouse plants (2).
- delete - no additional information is needed (3).
- delete - rules should not require analysis.
- delete - addressed in FSAR.

Item No. (2) (x)

(x) If determined necessary as a result of the analysis required by paragraph f(1)(XLi) of this section, an automatic power-operated relief valve isolation system shall be installed that will automatically cause the block valve to close when the reactor coolant system pressure falls after the PORV has opened (Applicable to PWRs only). (This requirement shall be implemented, if found to be necessary, by the end of the first refueling six months after staff approval of the design.) (II.K.3.1)

Public Comment

- delete - no acceptance criteria exists for this rule.
- Reference should be to (f)(1)(xxxi) vice (f)(1)(xli) (2).
- delete - under review by NRC staff.

Item No. (2) (xi)

(xi) The automatic depressurization system, valves, accumulators and associated equipment instrumentation shall show to be capable of performing their intended safety functions during and following exposure to the hostile environment of an accident situation, taking no credit for non-safety related equipment or instrumentation, and taking account for air (or nitrogen) leakage through valves. (Applicable to BWRs only.) (January 1, 1982) (II.K.3.28)

Public Comment

o delete - covered in 10 CFR 50.46 Part b(5). (6)

Item No. (2) (xii)

(xii) Plant-specific calculations for small break loss of coolant accidents shall be provided consistent with the revised models development pursuant to item f(1)(xLiii) of this section. (January 1, 1983) (II.K.3.31)

Public Comments

- o delete - already covered by 10 CFR 50.46 (6).
- o delete - rules should not require analysis.
- o delete - addressed in FSAR.

Item No. (2) (xiii)

(xiii) The RCIC system shall automatically transfer its suction to the suppression pool when the condensate storage tank level is low. (Applicable to BWRs only.) (January 1, 1982) (II.K.3.22)

Public Comments

- should reference Item II.K.3.22.b in NUREG-0737 not II.K.3.22 in toto.

Item No. (2) (xiv)

(xiv) The design of the automatic depressurization system shall be such that any operation of this system needed to assure adequate core cooling will be initiated automatically. The design description shall be submitted to the NRC for approval by April 1, 1982, or as part of the FSAR, whichever is later. For operating licenses issued prior to April 1, 1983, the design shall be installed not later than the first refueling outage that is at least six months subsequent to staff approval of the design. For operating licenses issued on or after April 1, 1983, the design shall be installed not later than the start of power operation (i.e., greater than 5% of rated power). (Applicable to BWRs only) (II.K.3.18)

Public Comments

- o Should reference Items II.D.3.18.b and c in NUREG-0737 not II.D.3.18 in toto.

Item No. (2) (xv)

(xv) The core spray and the LPCI systems shall automatically restart upon low water level, if an initiation signal is still present, to assure adequate core cooling. The design description shall be su mitted to the NRC for approval. For operating licenses issued prior to January 1, 1982 the design shall be installed not later than the first refueling outage that is at least six months subsequent to staff approval of the design. For operating licenses issued on or after January 1, 1982, the design shall be installed not later than the start of power operations (i.e., greater than 5% rated power). (Applicable to BWRs only.) (II.K.3.21)

Public Comments

- delete - BWR Owner's Group has concluded that modification would not enhance public safety (3).