January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-400-LA
	)	
(Shearon Harris Nuclear Power Plant)	)	ASLBP No. 99-762-02-LA

EXHIBITS SUPPORTING THE SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT

VOLUME 7

EXHIBIT 12 - 20

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NUCLEAR REGULATORY COMMISSION

In the Matter of CAROLINA POWER & LIGHT COMPANY (Shearon Harris Nuclear Power Plant)	Docket No. 50-400-LA ASLBP No. 99-762-02-LA

DEPOSITION OF:	LAURENCE I. KOPP, Ph.D.
DATE:	November 4, 1999 Commencing at 2:15 p.m.
PLACE:	Goya Conference Room Four Points Sheraton Hotel 37611 U.S. Highway 19 North Palm Harbor, Florida 34684
REPORTED BY:	Dale E. DeFranco, RPR Notary Public State of Florida at large

COPY

APPEARANCES:

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* E X H I B I T S *

DR. LAURENCE I. KOPP,

the deponent herein, being first duly sworn, was examined and testified as follows:

DIRECT EXAMINATION

BY MS. CURRAN:

Q. Good afternoon, Dr. Kopp.

A. Good afternoon.

Q. Have you been deposed before?

A. No, I haven't.

Q. I'm going to ask you questions regarding your involvement in the review of the Harris Spent Fuel Pool Expansion License Amendment Application.

A. Uh-huh.

Q. And your experiences regarding criticality analysis in general. I don't mean for my questions to be confusing and if there's any question that you don't understand please ask me for clarification and I'll do my best to clarify it.

If for any reason you need a break just ask for a break and you're welcome to one. You're under -- you're aware you're under oath --

A. Yes.

Q. -- in this deposition. Okay.

MS. CURRAN: I'd like to ask the court reporter to mark as Exhibit 7 the resume of Laurence, 14

L-a-u-r-e-n-c-e, C. Kopp, K-o-p-p, Senior Reactor Engineer.

(Whereupon, Exhibit Number 7 was marked for identification.)

Q. Dr. Kopp, is this is correct copy of your resume?

A. Yes.

Q. Do I understand correctly from your resume that you've been employed with the USNRC as a senior reactor engineer since 1965?

A. Well, I been employed with them since 1965, but not as a senior reactor engineer.

Q. How long have you been a senior reactor engineer?

A. Oh, probably eight or ten years.

Q. You have been identified by the NRC staff as the only expert witness on issues of criticality safety; are you aware of that?

A. No.

Q. Are you the only person at the NRC who is responsible for final approval of criticality safety applications?

A. In spent fuel pools?

Q. Yes.

A. Primarily there are some others and there are a few that are being trained in case I ever decide to retire.

Q. Would you say that you are familiar with all of the criticality analyses for spent fuel pools that are submitted to the NRC?

A. I would say yes, the ones that have been submitted since the early 1980ies. I think 1983, when I first started working on criticality analysis, spent fuel pools.

Q. Before 1983 were you involved in criticality analysis at all?

A. Not for spent fuel pools, no.

Q. Are you the person who is presently responsible for review of the criticality issues raised by the Harris License Amendment Application with respect to Pools C and D?

A. Yes.

Q. Now, the NRC staff issued request for additional information in April of 1999 which was responded to by CP&L in June of 1999 regarding criticality issues; is that correct?

A. I'm not sure of the dates, but yes, it requests for additional information around that time frame.

Q. And it was answered, right?

A. Yes.

Q. As far as you're concerned is the NRC's review of criticality issues with respect to Harris completed?

A. Essentially it is, yes.

Q. Have you reached a conclusion regarding the adequacy of the criticality analysis provided by CP&L?

A. Yes.

Q. And what is it?

A. I believe that it's adequate and meets our current regulations and requirements as far as criticality concerns.

Q. You were saying that you began working on criticality issues around 1983; is that correct?

A. Yes. Somewhere around there.

Q. Are you familiar with the history of criticality analyses preceding 1993?

MS. UTTAL: Objection. Criticality analysis relating to the entire spectrum?

Q. I'm sorry. The spent fuel pools.

A. Not very much.

Q. But somewhat?

A. Somewhat.

Q. I'm going to ask you some questions about that and if you don't know just tell me. Is it correct to say that around the time of issuance of General Design Criterion 62 to 10 CFR Part 50, Appendix A, there were no license tec specs requiring boron to be present in fuel pool cooling water?

A. Could you restate that.

Q. Is it correct to say that at the time, around the time of the promulgation of GDC 62 --

A. Uh-huh.

Q. -- there were no tec specs in nuclear power plant licenses requiring soluble boron to be present in fuel pool cooling water?

A. I'm not sure but I believe there always has been tec specs in many plants that are requiring soluble boron in the pool with a minimum concentration.

Q. Do you know if it is required by the tec specs for Harris?

A. There is a tec spec for Harris that requires a minimum boron concentration in the reactor cooling system and in portions of the refueling cancel, and during
15
refueling everything is connected to the spent fuel pool. So in that sense there is a specification for a minimum boron concentration for spent fuel pool. During refueling everything is open so there's a continuous system.

Q. So is it correct to say that the tec specs do not specifically call for a specific, for a level boron concentration in the spent fuel pools?

A. That's right.

Q. Do you know when was the first time that such a requirement was entered into the tec specs for any nuclear plant?

MS. UTTAL: By such a requirement --

Q. For soluble boron and spent fuel pools.

A. I don't recall.

Q. You haven't hazarded a guess or a arange?

A. As far as, as long as I remember it it's been in most of the tecnical specifications.

Q. But your memory goes back to, say, the early '80ies.

A. For that specification, yes.

Q. So why is Harris different?

A. Well, it has been in some specs and it has not been others. There was no requirement to have it in all the technical specifications up until a few years ago when we developed the improved technical specifications.

Q. What year was that?

A. I'd say about, between three to five years ago.

Q. Were the improved technical specifications --

A. Can I clarify?

Q. Sure.

A. I don't know if I said it was required in all technical specifications in the past. It had been in many technical specifications in the past but it was not a requirement as far as I know.

Q. Okay. Would this be a requirement in, when you talk about the new and improved technical specifications, would you be referring to New Reg 1431, the Standard Technical Specifications for Westingtonhouse Plants?

A. Yes.

Q. Now, my copy of that is dated September 1992. That's for Volume, it just says Volume I. It doesn't give a revision number. Revision Zero. Would that seem to be --

MR. HOLLAWAY: Do you want to take a look at that?

Q. I don't have the whole thing but you're welcome to look at it, the first page.

A. Yes. This is it. This is what I'm referring to.

Q. So that would have been around 1992.

A. Has it that long ago?

Q. Time flies, doesn't it? So just to make sure I understand, it was with the promulgation of these standard tec specs that it became an NRC requirement to include a provision for soluble boron in spent fuel pools in the tec specs.

A. For those plants that adopted the improved

15 standard tec specs and eliminated their older tec specs.

Q. And in what proportions of plants did that?

A. Well, it's a continuing basis. They're coming in gradually. Some of them are adopting the improved tec specs and others are not. I don't know what percentage have adopted them and what percentage have not adopted them.

Q. Is it the NRC's goal that most or all plants will adopt these?

A. I don't know if it's our goal, but it's a joint effort by NRC and industry to simply tec specs and make them more consistent among different venders.

Q. Would it be correct to say that at the time of the promulgation of GDC-62 or thereabouts, which was in 1971, there were no technical specifications in nuclear power plant licenses that imposed burnup limits on fuel to be stored in fuel pools?

A. As far as I recall that's right, there were none.

Q. When was a burnup limit on fuel stored in pools first inserted in a set of nuclear power plant tec specs?

A. I'd have to be guessing, but I will say the early 1980ies, maybe 1981, '82.

Q. Do you remember what the plant was?

A. No, I don't.

Q. Do you remember what the circumstances were?

A. I'm not sure.

Q. Was it a license amendment application for high density storage racks?

A. I'm not sure if it was a license amendment or just a vender that came in requesting NRC review and approval of the concept.

Q. And the NRC approved the concept?

A. Yes.

Q. Do you know whether it was the plant's specific approval or a generic approval that was given?

A. I'm not sure.

Q. Do you know if at the time the NRC gave approval of the change in the tec specs was this accompanied by a criticality analysis?

A. Yes. I would say so. Although I was not involved in the review.

16 Q. At that time was there any supporting assessment by the NRC or the applicant or what ever vender was involved of the risk of making this change to the means for controlling criticality?

A. I don't know. I was not involved.

Q. Has the NRC performed any evaluations of the likelihood of a boron dilution event in Pools C and D at Harris?

A. The NRC, no, no.

Q. Has the NRC requested CP&L to perform any such evaluation?

A. The evaluation was performed. It was part of the license amendment. The criticality analysis that shows there's a five percent criticality margin without boron in the pool water is essentially the boron dilution event.

Q. But was there any evaluation by CP&L or any other party of the likelihood of a boron dilution event in Pools C and D?

A. Not that I know of, no.

Q. Has the NRC performed any evaluation of the set of scenarios by which CP&L might place one or more fuel assembly -- strike that. Has the NRC performed any evaluation of the set of scenarios by which CP&L might place one or more out-of-compliance fuel assembly in Pools C or D?

MS. UTTAL: Do you need the question to be more specific?

THE DEPONENT: Yeah. I'm not sure of the evaluation of the scenario.

Q. Has the NRC attempted to identify possible scenarios or predict the probability of scenarios by which CP&L might place one or more out-of-compliance fuel assemblies in Pools C or D?

MS. UTTAL: Objection. That's a compound question. You're asking him whether the NRC has done analysis to predict the probability or I assume an analysis of the, how something like that would happen? Are those the two things you're asking?

Q. To identify possible scenarios and to predict their probability. Or to predict their probability?

A. No. We haven't done either of those. If I may say, we assumed the probability of misplacing the fuel assemblies, one, that's why we required the analysis be done, for 100 percent probability to misplace the fuel assembly. That was the basis for our request for additional information on that analysis.

Q. Has the NRC done any evaluation of the set of scenarios by which through a single error CP&L might place more than one out-of-compliance fuel assembly in Pools C or D?

A. No.

Q. Has the NRC requested CP&L to perform any such analysis?

A. No.

Q. I'm going to ask you a question that may sound familiar. I asked Dr. Turner the same thing. I'd like to ask some questions regarding how you, as a professional criticality analyst, would evaluate the envelope of criticality events -- I'm sorry, to identify the envelope of criticality events that could occur at Harris.

I'd like you to consider a hypothetical problem in criticality analysis for fuel in pools. In this problem the physical configuration of the racks is fixed. The variables are, one, soluble boron and, two, the combined burnup, slash, enrichment of the fuel. In this problem some number of fuel assemblies may exceed acceptable burnup, splash, enrichment levels.

In addition boron concentration may be anywhere from zero to 2000 PPM. Your task in this problem is to identify the set of scenarios involving combinations of parameters, one, which is soluble boron and, two, which is the combined burnup, slash, enrichment of the fuel. Such that criticality just occurs. I.e., k effective equals one.

For the purposes of this problem I'd like you to define that set of scenarios as the envelope of criticality events for this pool.

A. Okay. First of all may I say that if the fuel assembly exceeds the burnup enrichment limits it's safer that it was below the burnup enrichment limits. The

16 curve is a curve that requires burnup to either meet that limit or exceed it. Not be below it. So if your burnup is higher than the tec spec limit on the burnup you're safer, you're fuel assembly is less reactive.

Q. And the problem I'm posing to you within the envelope the fuel will be more reactive; outside the envelope the fuel will be less reactive. Do you understand?

A. Right.

Q. Using this hypothetical how would one determine the envelope of scenarios involved in criticality?

17 A. We base it on NRC requirements which require criticality not to be reached but to maintain the five-percent subcriticality margin at all times, even for the worst conceivable accident which would be a loss of all the boron in pool water, somehow diluted the pool from 2000 PPM down to zero, the calculations showed you're still at least five-percent subcritical. And as far as an envelope of calculations that seems to me to be the bounding point.

Q. But I've asked you to look at a combination of events involving two factors.

A. Right.

Q. We have factor one, which is soluble boron and factor two, the combined burnup enrichment of the fuel.

So I'm asking you to look at combinations of these factors. And in looking at those combinations I would assume there are various combinations you would look at, how would you determine the envelope?

MS. UTTAL: Objection. He's already stated that the envelope or the bounding, the boundary is the loss of boron. I don't know what purpose is served by talking about things that are inside the boundaries, or he stated what the outside boundary is.

A. The burnup curves are based on a five-percent subcriticality margin assuming no credit for boron. One never goes through some type of analysis where you would be critical, so one never does calculate a k effective 1.0. We always have a five-percent safety margin.

Q. But what you just said to me assumes that there's only one misplaced fuel assembly, right?

A. No. What I said assumes that we have the maximum reactivity accident possible. Loss of all the boron in the pool water.

Q. Is that your professional answer as a scientist to the question?

A. Yes.

Q. One of the pieces of guidance that the NRC uses to evaluate criticality analyses is proposed Revision 2 to Reg Guide 1.13 which is dated December 1981; is that correct?

Would you like me to show you that document?

A. I know what document you're talking about. That was never officially issued as a reg guide and therefore it's hard to say that it's something that the commission would rely on.

Q. Well, I'm asking what the staff relies on. Does the staff rely on it?

A. I do not rely on it. I know what's in there, but I know some of things in there are not, are obsolete and I know some of the things in there have been updated since then.

Q. In terms of criticality analysis. Well, why don't I pass this out and ask the reporter to mark as Exhibit 8, Proposed Revision 2 to Regulatory Guide 1.13, entitled Spent Fuel Storage Facility Design Basis. It's dated December 1981.

(Whereupon, Exhibit Number 8 was marked for identification.)

Q. Dr. Kopp, this document that I'm showing you, is this a copy of the guidance document we've been talking about?

A. It's a copy of Proposed Reg Guide 1.13, yes.

Q. Okay. Now as you were saying a little earlier that is still in proposed or draft form, right?

A. I don't know what's happened to it now. It's almost 20 years and I have not heard anything about it.

Q. You haven't heard any rumors that it's about to come out?

A. No.

Q. We're all waiting. Can you tell me what measures or what aspects of the guidance that relate to criticality control have been changed or updated?

A. As far as I see it's been updated to reflect the recent position that we have granted in partial credit for soluble boron for normal conditions.

Q. And in what case was that granted?

A. It was a generic topical report from Westinghouse that we reviewed and approved three years ago. Somewhere in that time frame.

Q. So that's the one thing that you can offer that is changed since this --

A. Yes.

Q. -- draft or reg guide was published?

A. From quickly glancing at it, yes.

Q. Okay. Dr. Kopp, you were here for the deposition of Dr. Turner; is that correct?

A. Yes.

Q. Do you recall a discussion about an

17 unsubstantiated rumor that the NRC is having second thoughts and may rescind it's approval of partial boron credit?

A. Yes.

Q. Is there any truth to that rumor?

A. Not that I know of.

Q. Do you know where that rumor may have come from?

A. No, I don't.

Q. You have no idea what would have started it?

A. No. We have granted partial boron credit for several plants already.

Q. Which plants are those?

18 A. Prairie Island was the first. I believe Vogtle was another one. V-o-g-t-l-e.

Q. Was Comanche Peak another one?

A. I'm not sure if it was Comanche Peak or South Texas. It might have been both of those. Some have not, the amendment has not officially gone out yet.

St. Lucy II was also another. I'm not sure which of these have already been officially approved and which were just approved by our branch as far as criticality goes.

MS. CURRAN: I'd like to as the court reporter to mark as Exhibit 9 an August 1998 memorandum from

Laurence Kopp, Senior Reactor Engineer, Reactor System Branch, Division of Systems Safety and Analysis to Timothy Collins, Chief Reactor Systems Branch, Division of Systems Safety and Analysis.

Attached to it is a document entitled Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants.

(Whereupon Exhibit Number 9 was marked for identification.)

Q. Do you recognize this document, Dr. Kopp?

A. Yes.

Q. To what extent, if any, did you participate in the preparation of the guidance document that's attached to this cover memo?

A. Well, I finalized it. It was begin maybe eight or ten years previous to this by several members of our branch. And we finally got it issued in 1998.

Q. So at the time this was written you approved it and it represented your views.

A. Yes. It was an update of the previous guidance that had gone out by the so-called Grimes letter, things that had been approved since then. And we wanted to get them all down in an official document and told what they were.

Q. Does that document supersede the Grimes letter?

A. Well, it's not an additional NRC document. It's a memo from me to my branch chief and it was put in the PDR, but as such it is not an official NRC document.

Q. But the Grimes letter is in contrast?

A. Well, the Grimes letter would be equivalent to a generic letter nowadays. I guess back in those times they didn't have such a thing. So the Grimes letter went through a series of compariances, various, I'm sure how high. But this only went from myself to my branch chief.

But it presents no new policy, it's just an update of existing methodologies and approvals that have been made by the staff since the so-called Grimes letter.

Q. So this document summarizes the state of existing regulatory guidance?

A. That's right.

Q. Would you please turn to page 7 of the attached guidance document. Do you see towards bottom of the page Section B entitled Additional Considerations?

A. Yes.

Q. Do you see also paragraph 2 under heading B?

A. Yes.

Q. Would you read paragraph 2 to yourself.

A. When this, there's a sentence in paragraph 2 that states, "Normally a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors creditible." Is that correct?

A. Yes.

Q. In evaluating a license amendment application or a licensing application for spent fuel storage that involves criticality analysis, does the NRC apply this particular sentence, this consideration raised in this sentence to the application?

A. It applies in that normally we consider, we require an analysis of of a single misloading event.

Q. Normally does the staff make any determination as to whether there are circumstances that's make multiple loading errors credible?

A. The staff doesn't. No.

Q. Does the staff ask a licensee or license applicant to do that?

A. Not that I'm aware of, no.

Q. Did the staff ask the license applicant to do that in the Harris case?

A. No.

Q. So is it fair to say that the staff simply doesn't apply the aspect of this sentence which says "unless there are circumstances that make multiple

18 loading errors credible?"

MS. UTTAL: Objection. You can answer.

A. The reason this is in here is that if something developes in the future, that I can't foresee now what the circumstance would be, but we wanted something in here to cover possible circumstances in the future where more than a single misloading might be feasible.

We have not run into that as far as I know to the present time, but to make this all encompassing instead of revising it for the future, we decided to put something like that to cover future possible circumstances.

Q. But the purpose of this is to provide for such consideration in case you should run into such
19 circumstances; is that right?

A. Yes.

Q. But you don't go looking for them, you just wait to see if you run into them.

A. Yes.

Q. Please describe what circumstances, if any, and under what regulatory requirements, if any, the NRC requires recording of the misplacement of fresh or spent fuel in spent fuel storage pools.

A. It would probably be a licensing, an LER. If you violated your tec spec requirements or fuel loading patterns there would be a license event report that is required.

Q. Okay. And in an earlier discussion I had asked you if you could give me a date when, I had asked you to give me a date when burnup limits were first included in tec specs, and what was the date you gave me?

A. I think it was early '80ies; 1982, 1983.

Q. So would that correspond to fuel loading requirements?

A. I'm trying to think of when we went to multi-region racks and whether it was, if it was only due to burnup credit or whether there may have been some other reason. I'm not sure, but I would think that would be around the time we first --

Q. To your knowledge does the NRC keep records, data, or documents that describe the practical experience of nuclear power plant operators with fresh or spent fuel misplacement in fuel storage pools?

A. Well, in the sense that these events are reported to LER's, there would be a record of them. Whether there's a compilation of them I'm not sure. But they're certainly available.

Q. For your purposes or for your divisions purposes of evaluating criticality analyses does the NRC keep any such compilations?

A. Not that I know of. Mainly for the reason that we require a misloading event to be analyzed anyway.

Q. I'm sorry, I didn't --

A. We require a misloading event to be one of the analyzed accidents. Whether there are incidents in the past or not does not seem to be of concern here since we require the event anyway to be analyzed.

Q. Do you see any difference between misplacement, misidentification and mischaracterization of spent fuel assemblies, or fuel assemblies?

A. Would you please -- misplacement --

Q. Misplacement, misidentification and mischaracterization.

A. It would seem to me it would be three separate entities, items.

Q. Has the NRC ever evaluated whether a single misidentification error could lead to multiple misplacement of fuel?

A. No.

Q. Has the NRC ever evaluated whether a single mischaracterization error could lead to multiple misplacement of fuel?

A. No. When you say the "NRC," I'm answering for myself. For my experience with this analysis. I don't recall any other offices in the NRC that would be evaluating this either. I can't speak for them.

Q. And are you familiar with any industry evaluations of either of those two things, either mischaracterization or misidentification leading to misplacement of more than one assembly?

A. No.

Q. Has the NRC performed or obtained any analysis or evaluation of the practical experience of nuclear power plant operators with fresh or spent fuel misplacement in fuel storage pools?

A. Obtained what?

Q. Do you want me to read it again?

A. Yes, please.

Q. Has the NRC performed or obtained any analysis or evaluation of the practical experience of nuclear power plants operators with fresh or spent fuel misplacement in fuel storage pools?

MS. UTTAL: Objection. Could you be more percise in your term about practical experience. Do you mean an analysis of actual events?

Q. By practical experience I mean actual events.

A. The question was have we evaluated any actual events of misloadings?

Q. Right. Well, have you evaluated the composite experience of licensees with misloading events?

19 A. No.

Q. Has the NRC performed or obtained any analysis of the probability of misplacing fresh or spent fuel in fuel storage pools?

A. As I said before, we assume the probabilities is a hundred percent because we require that analysis to be performed. We require a misplaced fuel assembly to be analyzed.

Q. But let me just clarify. You require as a matter of practice the misplacement of a single fuel assembly to be analyzed.

A. Right.

Q. My question was broader than that.

A. Could you repeat it.

Q. Has the NRC performed or obtained any analysis
20 of the probability of misplacing fresh or spent fuel in fuel storage pools?

Why don't I try answering and you tell me if my answer is correct.

A. Yeah. Throwing in the probabilities is what is confusing me. As I said, by requiring the event to be analyzed we assume its probability is a hundred percent in a single misplaced assembly.

Q. And you haven't looked at the probability of misplacing more than one fuel assembly.

A. We haven't looked at the probability, but we have done analysis of misplacing more than one fuel assembly.

Q. Could you explain that.

A. Well, we had someone in our branch recently, it wasn't Dr. Thompson's concern, misplace fresh fuel assembly in every location in a rack, in one of the Shearon Harris racks. And I think he was conservative in that he used a 10 by 8 arrangement of racks which would be 80 cells. So 80 cells contain fresh five weight-percent fuel. And the result was still less than critical. That was 200 PPM of boron.

So misplacing 80 fuel assemblies is highly conservative because there probably wouldn't be anywhere near that many fresh pool assemblies at a time on site and misloading a whole rack is highly unlikely.

Q. Was this analysis documented anywhere?

A. No. We just performed it recently. Within a week or so.

Q. So, but did someone write it down when you performed it?

A. No. It's in the process now of being officially documented.

Q. Will we be able to get a copy of it when it is?

A. Sure.

Q. Okay. You just described for me an analysis that the staff did assuming that there were 80 misplaced fuel assemblies in Pools C or D.

A. Yes.

Q. And that the boron was present --

A. Yes.

Q. -- at 2000 PPM.

A. Right.

Q. And you had told me earlier that at the other extreme you have evaluated a situation where there's no fuel misplacement but there's no boron.

A. That's right.

Q. And you've also evaluated, or Holtec has evaluated, a situation where there is one fuel misplacement and no boron.

A. That's right. Which is beyond what we normally require for analysis.

Q. So is it correct to say that those are basically two extremes?

A. I would say so, yes.

Q. Are there other combinations of events that you could envision that could cause criticality; for instance, misplacement of less, somewhere between zero and 80, or one in 80 fuel assemblies, and some diminution in boron concentration in the pool?

A. I don't know. I haven't looked at that. I haven't considered that because that does not conform to what we, our basis for the double contingency principle.

Q. But would it be correct to say that if you as a scientist were trying to establish the envelope of scenarios that could cause criticality at Harris, that you would look at various combinations of the two factors, fuel misplacement and boron dilution, to see where in those various combinations criticality could occur?

MS. UTTAL: That's been asked and answered.

Q. Please answer the question.

A. No. We don't look at those scenarios.

Q. I'm not asking what you do look at, I'm asking what you would look at.

A. No. We would not look at that because we would consider scenarios like that to be highly, extremely unlikely.

Q. So why did you look at the misplacement of 80 fresh assemblies in the rack?

A. Just to satisfy ourselves and Dr. Thompson.

Q. Would it surprise you if Dr. Thompson weren't satisfied by that?

A. No.

MS. UTTAL: Diane, it's been about an hour, how about a break.

MS. CURRAN: Okay. Sure.

(Whereupon a break was taken.).

THE DEPONENT: May I clarify something?

MS. CURRAN: Sure.

THE DEPONENT: We talked about the memo from myself to Mr. Collins of August 19th, 1998.

Q. Yes.

A. It was not an official NRC document. I would like to clarify the reason for the document. It was primarily for guidance for new members or future members of our branch that would be doing criticality analysis of spent fuel pools. And after it was written the management decided that maybe it should be promulgated to industry too, so he put it in PDR as an updated, essentially, version of the Grimes letter.

So original reason it was written was for members of our branch, new members of our branch that might be coming in to supply them with things that we have already reviewed and separated as far as guidance in the criticality analysis of spent fuel pools.

Q. When was it put in it PDR?

A. I guess August 19th, 1998.

Q. And did the staff solicit comments on the memo?

A. No. Within the branch.

Q. Oh, within the branch --
A. Right.
Q. -- you solicited comments, but not from the industry.
A. No. No.
Q. So you put it in the PDR with thoughts that you might get comments from industry if you put it there or, maybe I misunderstood you. When you said something about getting comments, you meant from within the branch?
A. Within the branch.
Q. And not from industry.
A. Right.
Q. Okay. In Dr. Turner's deposition he mentioned that he thought there might be a list of fuel misplacement events kept by the the NRC. Do you recall that?
A. As I said, there are LER's that come in, but I don't know if there's a compilation of them or not.
Q. So you're not?
A. I'm not familiar with them.
Q. Okay. Has the NRC performed any analysis of the consequence of misplacing one or more fresh or spent fuel assemblies in fuel storage pools?
A. Yes.
Q. And where would that be found?

A. The analysis I mentioned that was done this past week or so that we're preparing a memo on, where we misplaced an entire rack with fresh fuel assemblies.

Q. All right. But that's the only one that you know of?

A. That we did ourselves, that we analyzed ourselves? Yes.

Q. Do you know of any that have been prepared by any other entity?

MS. UTTAL: Are you speaking of CP&L's application or any application?

Q. Anyone.

MS. UTTAL: Any kind of analysis of misplacement.

A. You mean where there have been multiple misplacements?

MS. UTTAL: She said single or more.

Q. Single or multiple.

A. I'm not certain. I have seen statements in the past that to the effect that an entire rack could be misloaded with fresh fuel assemblies and with credit for boron, one would still maintain the five-percent subcriticality margin. I have seen that with various other submittals.

But we've never varified that ourselves. and that's why we decided this week to actually do a calculation and see if would be true for Shearon Harris. And we found we are subcritical for the entire rack.

Q. Okay. Under what circumstances, if any, and under what regulatory requirements, if any, does the NRC require the reporting of errors in controlling boron concentration in the water of fuel storage pools?

A. I'm not sure if there would be any requirements for reporting that. If the boron concentration were a minimum boron concentration were in tec specs and if that were violated during the surveillance interval, there would be a certain amount of time where one could reborate and get back up to the required minimum level. And that would not be really I guess reportable unless one did not borate in time. There's a certain interval where you come back within regulations.

A. I see. And if you correct it with appropriate intervals it's not a reportable event; is that what you're saying?

A. Right.

Q. Okay. To the extent that boron dilution events are reported to the NRC, does the NRC keep any centralized record of boron dilution events that you know?

A. It would be the same as the LER's for fuel misplacements. There would be the LER's as far as I know. We don't compile them but they're available.

Q. Has the NRC performed or obtained any analysis or evaluation of nuclear power plant operator's experience with controlling boron concentrations in fuel storage pools?

A. Not that I know of.

MS. CURRAN: I'd like to ask the court reporter to mark as Exhibit 10 an October 25th, 1996 letter from Timothy E. Collins, Acting Chief, Reactor System Branch, Division of System Safety and Analysis, NRC, to Mr. Tom Green, Chairman Westinghouse Owner's Group. Subject: Acceptance for Referencing of Licensing Topical Report WCAP-14416-P, Westinghouse Special Fuel Rack Criticality Analysis Methodology.

Attached to this cover letter is a Safety Evaluation by the Office of Nuclear Reactor Regulation relating to Topical Report WCAP-14416-P.

(Whereupon, Exhibit Number 10 was marked for identification.)

Q. Dr. Kopp, are you familiar with this document?

A. Yes, I am.

Q. If you would turn to page 10 -- actually page 10 is a continuation of a discussion that starts on page

8, Section 3.7 entitled Soluble Boron Credit Methodology; isn't that correct?

A. Yes.

Q. If you look at the second full paragraph on page 10 of the SER, I'd like to ask you about a sentence that reads: "However, a boron dilution analysis will be performed for each plant requesting soluble boron credit to ensure that sufficient time is available to detect and mitigate the dilution before the 0.95 k effective design basis is exceeded and submitted to the NRC for review." In parentheses, "Ref, dot, 29."

Can you explain to me what is meant by this sentence and the reference to Ref 29?

A. Yes. This is the new methodology that I spoke of earlier. This is one of the reasons for updating the Grimes letter. This is a recent approval we gave for crediting partial soluble boron in spent fuel pools. And since we are allowing, not for Shearon Harris, but for some reactors, credit for soluble boron under normal conditions to meet .95, this would now require a new accident to be evaluated which would be the boron dilution event.

For other plants, such as Shearon Harris, which do not take credit for soluble boron during normal conditions, the fact that they calculate the five percent subcriticality margin in pure water takes care of the boron dilution event, that is complete dilution.

For these newer plants that want to take credit for the new methodology. They still must show they are subcritical with no boron, k effective is less than one, but to meet the k arc criteria, k effective less than or equal to .95, they can take credit for a certain amount of soluble boron. So because of that we require them now to do a boron solution analysis to show that they would get them below .95 dilution event.

Q. Okay. But Reference 29 in parentheses, when I turn to the back of this SER, Reference 29 is "Cassidy, B., et. al., Westinghouse Owners Group Evaluation of the Potential for Diluting PWR Spent Fuel Pools, WCAP-14181, July 1995."

How does that Reference 29 relate to what we were just reading on page 10?

A. That was a companion to this Westinghouse report which requested credit for partial boron. In order to prove that methodology I said they have to do a boron dilution event analysis. And this other report that you referenced shows how to do an analysis of a boron dilution event in the PWR.

Q. So the reason for the mention of Reference 29 is that this is a way for licensees to do the boron dilution analysis and that, that will meet NRC approval?

A. When they want credit for this methodology, partial boron credit, yes.

Q. And has the NRC approved Reference 29 for that purpose?

A. No. The approval of a boron dilution event we decide is done on a case by case basis because the plans vary so much. The amount of, the volume of water that can be inserted into a pool for dilution varies from plant to plant through the mode of inserting it, the capacity of the pools vary. We decided a generic dilution event would not be worth anything or worth much, so we decided to, the people that wanted to accept this methodology for partial boron credit would have to do a plan specific for boron dilution analysis for their specific spent fuel pool. That's why that boron dilution event was never approved or accepted. It was a generic type of topical report.

Q. Okay.

Q. Has the NRC performed or obtained any analysis of the probability and/or consequences of potential accidents resulting from improper boron concentration in fuel storage pool water?

A. Only the analysis that shows that the zero PPM of boron when there's still a five-percent subcritical complete dilution.

Q. That was the analysis you referred to earlier, right?

A. That's the analysis that everyone does, is required to do, for their spent fuel pool except those that want to adopt a new methodology for partial boron credit.

Q. Has the NRC performed or obtain any analysis of the probability and/or consequences of potential criticality events in spent fuel storage pools involving fresh and/or spent fuel pool?

A. The single fuel assembly misloading event that is normally presented and was presented by Shearon Harris for their cooling, plus the recent calculation that the NRC staff did in misloading an entire fuel rack with fresh fuel.

MS. CURRAN: I don't have anymore questions.

MR. HOLLAWAY: I have one question.

CROSS-EXAMINATION

BY MR. HOLLAWAY:

Q. Dr. Kopp, this should be simple. Just one question. Dr. Kopp, in your opinion does the term "reactivity" include the effects of burnup?

A. Certainly burnup determines the reactivity of a fuel assembly.

MR. HOLLAWAY: That's all. I have no other questions.

MS. UTTAL: I have no questions.

MS. CURRAN: Okay.

* * * * *

(Whereupon, these proceedings concluded at 3:30 p.m.)

STATE OF FLORIDA)

COUNTY OF PINELLAS)

I, Dale E. DeFranco, do hereby certify that the deposition of DR. LAURENCE I. KOPP was held in the Matter of Carolina Power & Light Company, at the time and place set forth in the caption hereof; that I was authorized to and did report in shorthand the testimony and proceedings had in said deposition, and that the foregoing pages, numbered 1 through 41, inclusive, constitute a true and correct transcription of my said shorthand report.

WITNESS MY HAND this 22th day of November, 1999, at St. Petersburg, Pinellas County, Florida.

Dale E. DeFranco

Court Reporter

14

NOTARY PUBLIC STATE OF FLORIDA
Dale E. DeFranco
Notary Public, State of Florida
Commission No. CC 570095
My Commission Exp. 07/17/2000
1-800-3-NOTARY - Fla. Notary Service & Bonding Co.

Laurence I. Kopp
Senior Reactor Engineer

Education

Ph.D., Nuclear Engineering, University of Maryland, 1968
M.S., Physics, Stevens Institute of Technology, 1959
B.S., Physics, Fairleigh Dickinson College, 1956.

Employment

U.S. Nuclear Regulatory Commission, Senior Reactor Engineer, 1965 - present
Performs safety evaluations of reactor license applications, technical specifications, core reloads, spent fuel storage facilities, and topical reports. Developed regulatory guides, information notices, generic letters, rulemaking related to reactor physics, safety analysis, and fuel storage. Assisted in development of improved technical specifications in areas of reactivity control, power distribution limits, and fuel storage.

Westinghouse Astronuclear Laboratory, Senior Scientist, 1963-1965
Evaluated nuclear analytical methods to be used in the design of NERVA rocket reactors. Analyzed experiments performed in the Los Alamos zero power reactor.

Martin-Marietta Nuclear Division, Senior Engineer, 1959-1963
Performed core physics calculations on fluidized bed and PM-1 reactors. Performed parametric studies of reactors applicable to nuclear rocket applications. Programmed several FORTRAN computer codes.

Federal Electric Corporation, Senior Programmer, 1957-1959

Curtiss-Wright Research Division, Programmer/physicist, 1956-1957

EXHIBIT
7

Professional Societies

American Nuclear Society
ANS-10 Mathematics and Computations Standards Committee
ANSI N-17 Standards Committee on Research Reactors, Reactor Physics & Radiation Shielding

Publications

"The NRC Activities Concerning Boraflex Use in Spent-Fuel Storage Racks," invited paper, American Nuclear Society Annual Meeting, June 1996.

"Potential Loss of Required Shutdown Margin During Refueling Operations," invited paper, American Nuclear Society Annual Meeting, June 1990.

"Recommended Programming Practices to Facilitate the Portability of Scientific Computer Programs," ANS Proceedings of the Topical Meeting on Computational Methods in Nuclear Engineering, April 1979.

"The Neutron Resonance Integral of Natural Dysprosium," Ph.D. thesis, University of Maryland, 1968.

"Pool Reactor Experiments with Control Rods," Transactions of the American Nuclear Society, Vol. 10, Pg. 16, 1967 (co-author).

"Procedures for Obtaining Few-Group Constants for Systems Having Rapid Spectral Variation With Position," Transactions of the American Nuclear Society, Vol. 8, pg. 303, 1965 (co-author).

"Improved Nuclear Design Method for NERVA Calculations - NSDM II, WANL-TME-1091, Westinghouse Astronuclear Laboratory, 1965 (co-author).

"Analysis of Experiments Performed in Los Alamos ZEPO Reactor," WANL-TME-273, Westinghouse Astronuclear Laboratory, 1963.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

FOR COMMENT

A. INTRODUCTION

General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. It also requires that these systems be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions. This guide describes a method acceptable to the NRC staff for implementing Criterion 61.

B. DISCUSSION

Working Group ANS-57.2 of the American Nuclear Society Subcommittee ANS-50 has developed a standard that details minimum design requirements for

*The substantial number of changes in this proposed revision has made it impractical to indicate the changes with lines in the margin.

This regulatory guide and the associated value/impact statement are being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. They have not received complete staff review and do not represent an official NRC staff position.

Public comments are being solicited on both drafts, the guide (including any implementation schedule) and the value/impact statement. Comments on the value/impact statement should be accompanied by supporting data. Comments on both drafts should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch, by MAR 5 1982

Requests for single copies of draft guides (which may be reproduced) or for placement on an automatic distribution list for single copies of future draft guides in specific divisions should be made in writing to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Technical Information and Document Control.

EXHIBIT

spent fuel storage facilities at nuclear power stations. This standard was approved by the American National Standards Committee N18, Nuclear Design Criteria. It was subsequently approved and designated ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations," by the American National Standards Institute on April 12, 1976.

Primary facility design objectives are:

a. To prevent loss of water from the fuel pool that would uncover fuel,

b. To protect the spent fuel from mechanical damage, and

c. To provide the capability for limiting the potential offsite exposures in the event of significant release of radioactivity from the fuel.

If spent fuel storage facilities are not provided with adequate protective features, radioactive materials could be released to the environment as a result of either loss of water from the storage pool or mechanical damage to fuel within the pool.

1. LOSS OF WATER FROM STORAGE POOL

Unless protective measures are taken to prevent the loss of water from a fuel storage pool, the spent fuel could overheat and cause damage to fuel cladding integrity, which could result in the release of radioactive materials to the environment. Equipment failures in systems connected to the pool could also result in the loss of pool water. A permanent coolant makeup system designed with suitable redundancy or backup would prevent the fuel from being uncovered should pool leaks occur. Further, early detection of pool leakage and fuel damage can be made using pool-water-level monitors and pool radiation monitors that alarm locally and also at a continuously manned location to ensure timely operation of building filtration systems. Natural events such as earthquakes or high winds can damage the fuel pool either directly or by the generation of missiles. Earthquakes or high winds could also cause structures or cranes to fall into the pool. Designing the facility to withstand these occurrences without significant loss of watertight integrity will alleviate these concerns.

2. MECHANICAL DAMAGE TO FUEL

The release of radioactive material from fuel may occur as a result of fuel-cladding failures or mechanical damage caused by the dropping of fuel elements or objects onto fuel elements during the refueling process and at other times.

Plant arrangements consider low-probability accidents such as the dropping of heavy loads (e.g., a 100-ton fuel cask) where such loads are positioned or moved in or over the spent fuel pool. It is desirable that cranes capable of carrying heavy loads be prevented from moving into the vicinity of the stored fuel.

Missiles generated by high winds also are a potential cause of mechanical damage to fuel. This concern can be eliminated by designing the fuel storage facility to preclude the possibility of the fuel being struck by missiles generated by high winds.

3. LIMITING POTENTIAL OFFSITE EXPOSURES

Mechanical damage to the fuel might cause significant offsite doses unless dose reduction features are provided. Dose reduction designs such as negative pressure in the fuel handling building during movement of spent fuel would prevent exfiltration and ensure that any activity released to the fuel handling building will be treated by an engineered safety feature (ESF) grade filtration system before release to the environment. Even if measures not described are used to maintain the desired negative pressure, small leaks from the building may still occur as a result of structural failure or other unforeseen events.

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protection requirements are acceptable for the water makeup source and its delivery system, the pool structure, the building housing the pool, and the filtration ventilation system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

Power Plants," provide guidelines to limit potential offsite exposures through the filtration-ventilation system of the pool building.

Occupational radiation exposure is kept as low as is reasonably achievable (ALARA) in all activities involving personnel, and efforts toward maintaining exposures ALARA are considered in the design, construction, and operational phases. Guidance on maintaining exposures ALARA is provided in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

C. REGULATORY POSITION

The requirements in ANSI N210-1976/ANS-57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations,"* are generally acceptable to the NRC staff as a means for complying with the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 as related to light-water reactors (LWRs), subject to the following clarifications and modifications:

1. In lieu of the example inventory in Section 4.2.4.3(1), the example inventory should be that inventory of radioactive materials that are predicted to leak under the postulated maximum damage conditions resulting from the dropping of a single spent fuel assembly onto a fully loaded spent fuel pool storage rack. Other assumptions in the analysis should be consistent with those given in Regulatory Guide 1.25 (Safety Guide 25), "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

2. In addition to meeting the requirements of Section 5.1.3, boiling of the pool water may be permitted only when the resulting thermal loads are properly accounted for in the design of the pool structure, the storage racks, and other safety-related structures, equipment, and systems.

*Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525

3. In addition to meeting the requirements of Section 5.1.3, the fuel storage pool should be designed (a) to prevent tornado winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and (b) to prevent missiles generated by tornado winds from striking the fuel. These requirements are discussed in Regulatory Guide 1.117, "Tornado Design Classification." The fuel storage building, including walls and roof, should be designed to prevent penetration by tornado-generated missiles or from seismic damage to ensure that nothing bypasses the ESF-grade filtration system in the containment building.

4. In addition to meeting the requirements of Section 5.1.5.1, provisions should be made to ensure that nonfuel components in fuel pools are handled below the minimum water shielding depth. A system should be provided that, either through the design or the system or through administrative procedures, would prohibit unknowing retrieval of these components.

5. In addition to meeting the requirements of Section 5.1.12.10, the maximum potential kinetic energy capable of being developed by any object handled above stored spent fuel, if dropped, should not exceed the kinetic energy of one fuel assembly and its associated handling tool when dropped from the height at which it is normally handled above the spent fuel pool storage racks.

6. In addition to meeting the requirements of Section 5.2.3.1, an interface should be provided between the cask venting system and the building ventilation system to minimize personnel exposure to the "vent-gas" generated from filling a dry loaded cask with water.

7. In addition to meeting the requirements of Section 5.3.3, radioactivity released during a Condition IV fuel handling accident should be either contained or removed by filtration so that the dose to an individual is less than the guidelines of 10 CFR Part 100. The calculated offsite dose to an individual from such an event should be well within the exposure guidelines of 10 CFR Part 100 using appropriately conservative analytical methods and assumptions. In order to ensure that released activity does not bypass the

filtration system, the ESF fuel storage building ventilation should provide and maintain a negative pressure of at least 3.2 mm (0.125 in.) water gauge within the fuel storage building.

8. In addition to the requirements of Section 6.3.1, overhead handling systems used to handle the spent fuel cask should be designed so tnat travel directly over the spent fuel storage pool or safety-related equipment is not possible. This shouid be verified by analysis to show that the physical structure under all cask handling pathways will be adequately designed so that unacceptable damage to the spent fuel storage facility or safety-related equipment will not occur in the event of a load drop.

9. In addition to the references listed in Section 6.4.4, Safety Class 3, Seismic Category I, and safety-related structures and equipment should be subjected to quality assurance programs that meet the applicable provisions of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Further, these programs should obtain guidance from Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)," endorsing ANSI N45.2, and from the applicable provisions of the ANSI N45.2-series standards endorsed by the following regulatory guides:

1.30 (Safety Guide 30) "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment" (N45.2.4).

1.38 "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants" (N45.2.2).

1.58 "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel" (N45.2.6).

1.64 "Quality Assurance Requirements for the Design of Nuclear Power Plants" (N45.2.11).

1.74 "Quality Assurance Terms and Definitions" (N45.2.10).

1.88 "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records" (N45.2.9).

1.94 "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants" (N45.2.5).

1.116 "Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems" (N45.2.8).

1.123 "Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants" (N45.2.13).

10. The spent fuel pool water temperatures stated in Section 6.6.1(2) exceed the limits recommended by the NRC staff. For the maximum heat load during Condition I occurrences with normal cooling systems in operation and assuming a single active failure, the pool water temperature should be kept at or below 60°C (140°F). Under abnormal maximum heat load conditions (full core unload) and also for Condition IV occurrences, the pool water temperature should be kept below boiling.

11. A nuclear criticality safety analysis should be performed in accordance with Appendix A to this guide for each system that involves the handling, transfer, or storage of spent fuel assemblies at LWR spent fuel storage facilities.

12. The spent fuel storage facility should be equipped with both electrical interlocks and mechanical stops to keep casks from being transported over the spent fuel pool.

13. Sections 6.4 and 9 of ANS-57.2 list those codes and standards referenced in ANS-57.2. Although this regulatory guide endorses with clarifications and modifications ANS-57.2, a blanket endorsement of those referenced codes and

standards is not intended. (Other regulatory guides may contain some such endorsements.)

D. IMPLEMENTATION

The purpose of this section is to provide information tc applicants regar ing the NRC staff's plans for using this regulatory guide.

This proposed revision has been released to encourage public participatio in its development. Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method to be described in the active guide reflecting public comments will be used in the evaluation of applications for construction permits and operating licenses docketed after the implementation date to be specified in the active guide. Implementation by the staff will in no case be earlier than June 30, 1982.

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

1.1 A nuclear criticality safety analysis should be performed for each system that involves the handling, transfer, or storage of spent fuel assemblies at light-water reactor (LWR) spent fuel storage facilities.

1.2 The nuclear criticality safety analysis should demonstrate that each LWR spent fuel storage facility system is subcritical (k_{eff} not to exceed 0.95).

1.3 The nuclear criticality safety analysis should include consideration of all credible normal and abnormal operating occurrences, including:

a. Accidental tipping or falling of a spent fuel assembly,
b. Accidental tipping or falling of a storage rack during transfer,
c. Misplacement of a spent fuel assembly,
d. Accumulation of solids containing fissile materials on the pool floor or at locations in the cooling water system,
e. Fuel drop accidents,
f. Stuck fuel assembly/crane uplifting forces,
g. Horizontal motion of fuel before complete removal from rack,
h. Placing a fuel assembly along the outside of rack, and
i. Objects that may fall onto the stored spent fuel assemblies.

1.4 At all locations in the LWR spent fuel storage facility where spent fuel is handled or stored, the nuclear criticality safety analysis should demonstrate that criticality could <u>not</u> occur without at least two unlikely, independent, and concurrent failures or operating limit violations.

1.5 The nuclear criticality safety analysis should explicitly identify spent fuel assembly characteristics upon which subcriticality in the LWR spent fuel storage facility depends.

1.6 The nuclear criticality safety analysis should explicitly identify design limits upon which subcriticality depends that require physical verification at the completion of fabrication or construction.

1.7 The nuclear criticality safety analysis should explicitly identify operating limits upon which subcriticality depends that require implementation in operating procedures.

2. CALCULATION METHODS AND CODES

Methods used to calculate subcriticality should be validated in accordance with Regulatory Guide 3.41, "Validation of Calculational Methods for Nuclear Criticality Safety," which endorses ANSI N16.9-1975.

3. METHOD TO ESTABLISH SUBCRITICALITY

3.1 The evaluated multiplication factor of fuel in the spent fuel storage racks, k_s, under normal and credible abnormal conditions should be equal to or less than an established maximum allowable multiplication factor, k_a; i.e.,

$$k_s \leq k_a$$

The factor, k_s, should be evaluated from the expression:

$$k_s = k_{sn} + \Delta k_{sb} + \Delta k_u + \Delta k_{sc}$$

where

k_{sn} = the computed effective multiplication factor; k_{sn} is calculated by the same methods used for benchmark experiments for design storage parameters when the racks are loaded with the most reactive fuel to be stored,

Δk_{sb} = the bias in the calculation procedure as obtained from the comparisons with experiments and including a , extrapolation to storage pool conditions,

Δk_u = the uncertainty in the benchmark experiments, and

Δk_{sc} = the combined uncertainties in the parameters listed in paragraph 3.2 below.

3.2 The combined uncertainties, Δk_{sc}, include:

a. Statistical uncertainty in the calculated result if a Monte Carlo calculation is used,

b. Uncertainty resulting from comparison with calculational and experimental results,

c. Uncertainty in the extrapolation from experiment to storage rack conditions, and

d. Uncertainties introduced by the considerations enumerated in paragraphs 4.3 and 4.4 below.

3.3 The various uncertainties may be combined statistically if they are independent. Correlated uncertainties should be combined additively.

3.4 All uncertainty values should be at the 95 percent probability level with a 95 percent confidence value.

3.5 For spent fuel storage pool, the value of k_a should be no greater than 0.95.

4. STORAGE RACK ANALYSIS ASSUMPTIONS

4.1 The spent fuel storage rack module design should be based on one of the following assumptions for the fuel:

a. The most reactive fuel assembly to be stored at the most reactive point in the assembly life, or

b. The most reactive fuel assembly to be stored based on a minimum confirmed burnup (see Section 6 of this appendix).

Both types of rack modules may be present in the same storage pool.

4.2 Determination of the most reactive spent fuel assembly includes consideration of the following parameters:

a. Maximum fissile fuel loading,
b. Fuel rod diameter,
c. Fuel rod cladding material and thickness,
d. Fuel pellet density,
e. Fuel rod pitch and total number of fuel rods within assembly,
f. Absence of fuel rods in certain locations, and
g. Burnable poison content.

4.3 The fuel assembly arrangement assumed in storage rack design should be the arrangement that results in the highest value of k_s considering:

a. Spacing between assemblies,
b. Moderation between assemblies, and
c. Fixed neutron absorbers between assemblies.

4.4 Determination of the spent fuel assembly arrangement with the highest value of k_s shall include consideration of the following:

a. Eccentricity of fuel bundle location within the racks and variations in spacing among adjacent bundles,
b. Dimensional tolerances,
c. Construction materials,
d. Fuel and moderator density (allowance for void formations and temperature of water between and within assemblies),

e. Presence of the remaining amount of fixed neutron absorbers in fuel assembly, and

f. Presence of structural material and fixed neutron absorber in cell walls between assemblies.

4.5 Fuel burnup determination should be made for fuel stored in racks where credit is taken for burnup. The following methods are acceptable:

a. A minimum allowable fuel assembly reactivity should be established, and a reactivity measurement should be performed to ensure that each assembly meets this criterion; or

b. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and a measurement should be performed to ensure that each fuel assembly meets the established criterion; or

c. A minimum fuel assembly burnup value should be established as determined by initial fuel assembly enrichment or other correlative parameters, and an analysis of each fuel assembly's exposure history should be performed to determine its burnup. The analyses should be performed under strict administrative control using approved written procedures. These procedures should provide for independent checks of each step of the analysis by a second qualified person using nuclear criticality safety assessment criteria described in paragraph 1.4 above.

The uncertainties in determining fuel assembly storage acceptance criteria should be considered in establishing storage rack reactivity, and auditable records should be kept of the method used to determine the fuel assembly storage acceptance criterion for as long as the fuel assemblies are stored in the racks.

Consideration should be given to the axial distribution of burnup in the fuel assembly, and a limit should be set on the length of the fuel assembly that is permitted to have a lower average burnup than the fuel assembly average.

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

5.1 Fixed neutron absorbers may be used for criticality control under the following conditions:

a. The effect of neutron-absorbing materials of construction or added fixed neutron-absorbers may be included in the evaluation if they are designed and fabricated so as to preclude inadvertent removal by mechanical or chemical action.

b. Fixed neutron absorbers should be an integral, nonremovable part of the storage rack.

c. When a fixed neutron absorber is used as the primary nuclear criticality safety control, there should be provision to:

(1) Initially confirm absorber presence in the storage rack, and

(2) Periodically verify continued presence of absorber.

5.2 The presence of a soluble neutron absorber in the pool water should not normally be used in the evaluation of k_s. However, when calculating the effects of Condition IV faults, realistic initial conditions (e.g., the presence of soluble boron) may be assumed for the fuel pool and fuel assemblies.

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

6.1 Consideration should be given to the fact that the reactivity of any given spent fuel assembly will depend on initial enrichment, ^{235}U depletion, amount of burnable poison, plutonium buildup and fission product burnable poison depletion, and the fact that the rates of depletion and plutonium and fission product buildup are not necessarily the same.

6.2 Consideration should be given to the practical implementation of the spei fuel screening process. Factors to be considered in choosing the screening method should include:

a. Accuracy of the method used to determine storage rack reactivity;
b. Reproducibility of the result, i.e., what is the uncertainty in tha result?
c. Simplicity of the procedure; i.e., how much disturbance to other operations is involved?
d. Accountability, i.e., ease and completeness of recordkeeping; and
e. Auditability.

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

Each nuclear power plant has a spent fuel storage facility. General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that fuel storage and handling systems be designed to ensure adequate safety under normal and postulated accident conditions. The proposed action would provide an acceptable method for implementing this criterion. This action would be an update of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis."

1.2 Need for Proposed Action

Since Regulatory Guide 1.13 was last published in December of 1975, additional guidance has been provided in the form of ANSI standards and NUREG reports. The Office of Nuclear Reactor Regulation has requested that this guide be updated.

1.3 Value/Impact of Proposed Action

1.3.1 NRC

The applicants' basis for the design of the spent fuel storage facility will be the same as that used by the staff in its review of a construction permit or operating license application. Therefore, there should be a minimum number of cases where the applicant and the staff radically disagree on the design criteria.

1.3.2 Government Agencies

Applicable only if the agency, such as TVA, is an applicant.

1.3.3 Industry

The value/impact on the applicant will be the same as for the NRC staff.

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

The guidance furnished on the design basis for the spent fuel storage facility should be updated.

2. TECHNICAL APPROACH

The American Nuclear Society published ANS-57.2 (ANSI N210), "Design Objective for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." Part of the update of Regulatory Guide 1.13 would be an evaluation of this standard and possible endorsement by the NRC. Also, recommendations made by Task A-36, which were published in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," would be included.

3. PROCEDURAL APPROACH

Since Regulatory Guide 1.13 already deals with the proposed action, logic dictates that this guide be updated.

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

Authority for this regulatory guide is derived from the safety requirements of the Atomic Energy Act of 1954, as amended, through the Commission's regulations, in particular, General Design Criterion 61 of Appendix A to 10 CFR Part 50.

4.2 Need for NEPA Assessment

The proposed action is not a major action as defined by paragraph 51.5(a)(10) of 10 CFR Part 51 and does not require an environmental impact statement.

5. CONCLUSION

Regulatory Guide 1.13 should be updated.

August 19, 1998

MEMORANDUM TO: Timothy Collins, Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Laurence Kopp, Sr. Reactor Engineer /s/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

Attached is a copy of guidance concerning regulatory requirements for criticality analysis of new and spent fuel storage at light-water reactor power plants used by the Reactor Systems Branch. The principal objective of this guidance is to clarify and document current and past NRC staff positions that may have been incompletely or ambiguously stated in safety evaluation reports or other NRC documents. It also describes and compiles, in a single document, NRC staff positions on more recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests. This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel.

Attachment:
As stated

DISTRIBUTION

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SRXB R/F	E. Kendrick
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H. Richings	L. Kopp
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EXHIBIT
3

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

GUIDANCE ON THE REGULATORY REQUIREMENTS FOR CRITICALITY ANALYSIS OF FUEL STORAGE AT LIGHT-WATER REACTOR POWER PLANTS

1. INTRODUCTION

This document defines the NRC Reactor Systems Branch guidance for the assurance of criticality safety in the storage of new (unirradiated or fresh) and spent (irradiated) fuel at light-water reactor (LWR) power stations. Safety analyses submitted in support of licensing actions should consider, among other things, normal operation, incidents, and postulated accidents that may occur in the course of handling, transferring, and storing fuel assemblies and should establish that an acceptable margin exists for the prevention of criticality under all credible conditions.

This guidance is not applicable to fuel storage in casks, nor does it consider the mechanical, chemical, thermal, radiological, and other aspects of the storage of new and spent fuel. The guidance considers only the criticality safety aspects of new and spent LWR fuel assemblies and of fuel that has been consolidated; that is, fuel with fuel rods reassembled in a more closely packed array.

The guidance stated here is based, in part, on (a) the criticality positions of Standard Review Plan (SRP) Section 9.1.1 (Ref. 1) and SRP 9.1.2 (Ref. 2), (b) a previous NRC position paper sent to all licensees (Ref. 3), and (c) past and present practices of the staff in its safety evaluation reports (SERs). The guidance also meets General Design Criterion 62 (Ref 4), which states:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations

The principal objective of this guidance is to clarify and document current and past staff positions that may have been incompletely or ambiguously stated in SERs or other staff documents. A second purpose is to state staff positions on recently proposed storage configurations and characteristics in spent fuel rerack or enrichment upgrade requests (for example, multiple-region spent fuel storage racks, checkerboard loading patterns for new and spent fuel storage, credit for burnup in the spent fuel to be stored, and credit for non-removable poison inserts). Although these statements are not new staff positions, this document compiles them in a single paper In addition, a recently approved staff position for pressurized-water reactors (PWRs) would allow partial credit for soluble boron in the pool water (Ref 5)

The guidance stated here is applicable to both PWRs and boiling-water reactors (BWRs) The most notable difference between PWR and BWR fuel storage facilities is the larger size of the fuel assemblies and the presence of soluble boron in the spent fuel pool water of PWRs

The determination of the effective multiplication factor, k_{eff}, for the new or spent fuel storage racks should consider and clearly identify the following:

a. fuel rod parameters, including:

   1. rod diameter

   2. cladding material and cladding thickness

   3. fuel rod pellet or stack density and initial uranium-235 (U-235) enrichment of each fuel rod in the assembly (a bounding enrichment is acceptable)

b. fuel assembly parameters, including:

   1. assembly length and planar dimensions

   2. fuel rod pitch

   3. total number of fuel rods in the assembly

   4. locations in the fuel assembly lattice that are empty or contain nonfuel material

   5. integral neutron absorber (burnable poison) content of various fuel rods and locations in fuel assembly

   6. structural materials (e.g., grids) that are an integral part of the fuel assembly

The criticality safety analysis should explicitly address the treatment of axial and planar variations of fuel assembly characteristics such as fuel enrichment and integral neutron absorber (burnable poison), if present (e.g., gadolinia in certain fuel rods of BWR and PWR assemblies or integral fuel burnable absorber (IFBA) coatings in certain fuel rods of PWR assemblies).

Whenever reactivity equivalencing (i.e., burnup credit or credit for imbedded burnable absorbers) is employed, or if a correlation with the reactivity of assemblies in a standard core geometry is used (k_∞), such as is typically done for BWR racks, the equivalent reactivities must be evaluated in the storage rack configuration. In this latter approach, sufficient uncertainty should be incorporated into the k_∞ limit to account for the reactivity effects of (1) nonuniform enrichment variation in the assembly, (2) uncertainty in the calculation of k_∞, and (3) uncertainty in average assembly enrichment.

If various locations in a storage rack are prohibited from containing any fuel, they should be physically or administratively blocked or restricted to non-fuel material. If the criticality safety of the storage racks relies on administrative procedures, these procedures should be explicitly identified and implemented in operating procedures and/or technical specification limits

2. CRITICALITY ANALYSIS METHODS AND COMPUTER CODES

A variety of methods may be used for criticality analyses provided the cross-section data and geometric capability of the analytical model accurately represent all important neutronic and geometrical aspects of the storage racks. In general, transport methods of analysis are necessary for acceptable results. Storage rack characteristics such as boron carbide (B_4C) particle size and thin layers of structural and neutron absorbing material (poisons) need to be carefully considered and accurately described in the analytical model. Where possible, the primary method of analysis should be verified by a second, independent method of analysis. Acceptable computer codes include, but are not necessarily limited to, the following:

- CASMO - a multigroup transport theory code in two dimensions
- NITAWL-KENO5a - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- PHOENIX-P - a multigroup transport theory code in two dimensions, using discrete ordinates
- MONK6B - a multigroup transport theory code in three dimensions, using the Monte Carlo technique
- DOT - a multigroup transport theory code in two dimensions, using discrete ordinates

Similarly, a variety of cross-section libraries is available. Acceptable cross-section libraries include the 27-group, 123-group, and 218-group libraries from the SCALE system developed by the Oak Ridge National Laboratory and the 8220-group United Kingdom Nuclear Data Library (UKNDL). However, empirical cross-section compilations, such as the Hansen-Roach library, are not acceptable for criticality safety analyses (see NRC Information Notice No. 91-26). Other computer codes and cross-section libraries may be acceptable provided they conform to the requirements of this position statement and are adequately benchmarked

The proposed analysis methods and neutron cross-section data should be benchmarked, by the analyst or organization performing the analysis, by comparison with critical experiments This qualifies both the ability of the analyst and the computer environment The critical experiments used for benchmarking should include, to the extent possible, configurations having neutronic and geometric characteristics as nearly comparable to those of the proposed storage facility as possible. The Babcock & Wilcox series of critical experiments (Ref 6) provides an acceptable basis for benchmarking storage racks with thin strong absorber panels for reactivity control. Similarly, the Babcock & Wilcox critical experiments on close-packed arrays of fuel (Ref. 7) provide an acceptable experimental basis for benchmark analyses for consolidated fuel arrays A comparison with methods of analysis of similar sophistication (e g transport theory) may be used to augment or extend the range of applicable critical experiment data

The benchmarking analyses should establish both a bias (defined as the mean difference between experiment and calculation) and an uncertainty of the mean with a one-sided tolerance factor for 95-percent probability at the 95-percent confidence level (Ref 8)

The maximum k_{eff} shall be evaluated from the following expression:

$$k_{eff} = k(calc) + \delta k(bias) + \delta k(uncert) + \delta k(burnup),$$

where

k(calc)	= calculated nominal value of k_{eff},
δk(bias)	= bias in criticality analysis methods,
δk(uncert)	= manufacturing and calculational uncertainties, and
δk(burnup)	= correction for the effect of the axial distribution in burnup, when credit for burnup is taken.

A bias that reduces the calculated value of k_{eff} should not be applied. Uncertainties should be determined for the proposed storage facilities and fuel assemblies to account for tolerances in the mechanical and material specifications. An acceptable method for determining the maximum reactivity may be either (1) a worst-case combination with mechanical and material conditions set to maximize k_{eff} or (2) a sensitivity study of the reactivity effects of tolerance variations. If used, a sensitivity study should include all possible significant variations (tolerances) in the material and mechanical specifications of the racks; the results may be combined statistically provided they are independent variations. Combinations of the two methods may also be used.

3. ABNORMAL CONDITIONS AND THE DOUBLE-CONTINGENCY PRINCIPLE

The criticality safety analysis should consider all credible incidents and postulated accidents. However, by virtue of the double-contingency principle, two unlikely independent and concurrent incidents or postulated accidents are beyond the scope of the required analysis. The double-contingency principle means that a realistic condition may be assumed for the criticality analysis in calculating the effects of incidents or postulated accidents. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Therefore, credit for the presence of the soluble boron may be assumed in evaluating other accident conditions.

4. NEW FUEL STORAGE FACILITY (VAULT)

Normally, fresh fuel is stored temporarily in racks in a dry environment (new fuel storage vault) pending transfer into the spent fuel pool and then into the reactor core However, moderator may be introduced into the vault under abnormal situations, such as flooding or the introduction of foam or water mist (for example, as a result of fire fighting operations). Foam or mist affects the neutron moderation in the array and can result in a peak in reactivity at low moderator density (called "optimum" moderation, Ref. 9). Therefore, criticality safety analyses must address two independent accident conditions, which should be incorporated into plant technical specifications:

a. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with pure water, the maximum k_{eff} shall be no greater than 0 95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

b. With the new fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with moderator at the (low) density corresponding to optimum moderation, the maximum k_{eff} shall be no greater than than 0.98, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

An evaluation need not be performed for the new fuel storage facility for racks flooded with low-density or full-density water if it can be clearly demonstrated that design features and/or administrative controls prevent such flooding.

Under the double-contingency principle, the accident conditions identified above are the principle conditions that require evaluation. The simultaneous occurrence of other accident conditions need not be considered.

Usually, the storage racks in the new fuel vault are designed with large lattice spacing sufficient to maintain a low reactivity under the accident condition of flooding. Specific calculations, however, are necessary to assure the limiting k_{eff} is maintained no greater than 0.95.

At low moderator density, the presence of relatively weak absorber material (for example, stainless steel plates or angle brackets) is often sufficient to preclude neutronic coupling between assemblies, and to significantly reduce the reactivity. For this reason, the phenomenon of low-density (optimum) moderation is not significant in racks in the *spent fuel pool* under the initial conditions before the pool is flooded.

Under low-density moderator conditions, neutron leakage is a very important consideration. The new fuel storage racks should be designed to contain the highest enrichment fuel assembly to be stored without taking credit for any nonintegral neutron absorber. In the evaluation of the new fuel vaults, fuel assembly and rack characteristics upon which subcriticality depends should be explicitly identified (e.g., fuel enrichment and the presence of steel plates or braces).

5. SPENT FUEL STORAGE RACKS

A. Reference Criticality Safety Analysis

1 For BWR pools or for PWR pools where no credit for soluble boron is taken, the criticality safety analyses must address the following condition, which should be incorporated into the plant technical specifications:

a With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than or equal to 0 95 including mechanical. and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level

2. If partial credit for soluble boron is taken, the criticality safety analyses for PWRs must address two independent conditions, which should be incorporated into the plant technical specifications:

   a. With the spent fuel storage racks loaded with fuel of the maximum permissible reactivity and flooded with full-density unborated water, the maximum k_{eff} shall be less than 1.0, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.

   b. With the spent fuel storage racks loaded with fuel of the maximum permissible activity and flooded with full density water borated to [*] ppm, the maximum k_{eff} shall be no greater than 0.95, including mechanical and calculational uncertainties, with a 95-percent probability at a 95-percent confidence level.[1]

3. The reference criticality safety analysis should also include, as a minimum, the following:

   a. If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable poison rods.

   b. For fuel assemblies containing burnable poison, the maximum reactivity should be the peak reactivity over burnup, usually when the burnable poison is nearly depleted.

   c. The spent fuel storage racks should be assumed to be infinite in the lateral dimension or to be surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of a reflector on the top and bottom of the fuel may be evaluated.

   d. The evaluation of normal storage should be done at the temperature (water density) corresponding to the highest reactivity. In poisoned racks, the highest reactivity will usually occur at a water density of 1.0 (i.e., at 4°C). However, if the temperature coefficient of reactivity is positive, the evaluation should be done at the highest temperature expected during normal operations; i.e., equilibrium temperature under normal refueling conditions (including full-core offload), with one coolant train out of service and the pool filled with spent fuel from previous reloads.

4. The fuel assembly arrangement assumed in the criticality safety analysis of the spent fuel storage racks should also consider the following

---

[1] [*] is the boron concentration required to maintain the 0.95k_{eff} limit without consideration of accidents

a. the effect of eccentric positioning of fuel assemblies within the storage cells

b. the reactivity consequence of including the flow channel in BWR fuel assemblies

5. If one or more separate regions are designated for the storage of spent fuel, with credit for the reactivity depletion due to fuel burnup, the following applies.

   a. The minimum required fuel burnup should be defined as a function of the initial nominal enrichment.

   b. The spent fuel storage rack should be evaluated with spent fuel at the highest reactivity following removal from the reactor (usually after the decay of xenon-135). Operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in storage cells of the designated region(s).

   c. Subsequent decay of longer-life nuclides, such as Pu-241, over the rack storage time may be accounted for to reduce the minimum burnup required to meet the reactivity requirements.

   d. A reactivity uncertainty due to uncertainty in the fuel depletion calculations should be developed and combined with other calculational uncertainties. In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption

   e. A correction for the effect of the axial distribution in burnup should be determined and, if positive, added to the reactivity calculated for uniform axial burnup distribution

B. Additional Considerations

1. The reactivity consequences of incidents and accidents such as (1) a fuel assembly drop and (2) placement of a fuel assembly on the outside and immediately adjacent to a rack must be evaluated. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these postulated accident conditions

2. If either credit for burnup is assumed or racks of different enrichment capability are in the same fuel pool, fuel assembly misloadings must be considered Normally, a misloading error involving only a single assembly need be considered unless there are circumstances that make multiple loading errors credible. Under the double-contingency principle, credit for soluble boron if present, is acceptable for these postulated accident conditions

3. The analysis must also consider the effect on criticality of natural events (e.g., earthquakes) that may deform, and change in the relative position of, the storage racks and fuel in the spent fuel pool.

4. Abnormal temperatures (above those normally expected) and the reactivity consequences of void formation (boiling) should be evaluated to consider the effect on criticality of loss of all cooling systems or coolant flow, unless the cooling system meets the single-failure criterion. Under the double-contingency principle, credit for soluble boron, if present, is acceptable for these abnormally elevated temperature conditions.

5. Normally, credit may only be taken for neutron absorbers that are an integral (nonremovable) part of a fuel assembly or the storage racks. Credit for added absorber (rods, plates, or other configurations) will be considered on a case-by-case basis, provided it can be clearly demonstrated that design features prevent the absorbers from being removed, either inadvertently or intentionally without unusual effort such as the necessity for special equipment maintained under positive administrative control.

6. If credit for soluble boron is taken, the minimum required pool boron concentration (typically, the refueling boron concentration) should be incorporated into the plant technical specifications or operating procedures A boron dilution analysis should be performed to ensure that sufficient time is available to detect and suppress the worst dilution event that can occur from the minimum technical specification boron concentration to the boron concentration required to maintain the $0.95k_{eff}$ design basis limit. The analysis should consider all possible dilution initiating events (including operator error), dilution sources, dilution flow rates, boration sources, instrumentation, administrative procedures, and piping. This analysis should justify the surveillance interval for verifying the technical specification minimum pool boron concentration.

7. Consolidated fuel assemblies usually result in low values of reactivity (undermoderated lattice). Nevertheless, criticality calculations, using an explicit geometric description (usually triangular pitch) or as near an explicit description as possible, should be performed to assure a k_{eff} less than 0.95.

6. REFERENCES

1. NRC, "Standard Review Plan" NUREG-0800, Rev 2, Section 9 1 1, "New Fuel Storage," July 1981.

2. NRC, "Standard Review Plan" NUREG-0800, Rev 2, Section 9 1 2, "Spent Fuel Storage," July 1981.

3. Brian K. Grimes, NRC, letter to all power reactor licensees, with enclosure "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978

4. *Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

5. Westingnouse Electric Corporation, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, November 1996.

6. Babcock & Wilcox Company, "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel," BAW-1484-7, July 1979.

7. Babcock & Wilcox Company, "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, 1981.

8. National Bureau of Standards, *Experimental Statistics*, Handbook 91, August 1963.

9. J. M. Cano, R. Caro, and J. M. Martinez Val, "Supercriticality Through Optimum Moderation in Nuclear Fuel Storage," *Nuclear Technology*, Volume 48, May 1980.

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

October 25, 1996

Mr. Tom Greene, Chairman
Westinghouse Owners Group
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230-0355

Dear Mr. Greene:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-14416-P, "WESTINGHOUSE SPENT FUEL RACK CRITICALITY ANALYSIS METHODOLOGY" (TAC NO. M93254)

The staff has reviewed the topical report submitted by the Westinghouse Owners Group by letter dated July 28, 1995, and supplemented by letter dated October 18, 1996. The report is acceptable for referencing in license applications to the extent specified and under the limitations stated in the enclosed U.S. Nuclear Regulatory Commission (NRC) evaluation. The evaluation defines the basis for acceptance of the report.

The staff will not repeat its review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented applies to the specific plant involved. NRC acceptance applies only to the matters described in the report. In accordance with procedures established in NUREG-0390, the NRC requests that the Westinghouse Owners Group publish accepted versions of the report, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed evaluation between the title page and the abstract and an -A (designating accepted) should follow the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion that the report is acceptable is invalidated, the Westinghouse Owners Group and/or the applicant referencing the topical report will be expected to revise and resubmit its respective documentation, or submit justification for the continued applicability of the topical report without revision of the respective documentation.

EXHIBIT
10
dcl 11/4/11

Sincerely,

Timothy E. Collins

Timothy E. Collins, Acting Chief
Reactor Systems Branch
Division of Systems Safety and Analysis

Enclosure:
WCAP-14416-P Evaluation

CPL 01920002

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

ENCLOSURE

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT WCAP-14416-P

"WESTINGHOUSE SPENT FUEL RACK CRITICALITY ANALYSIS METHODOLOGY"

WESTINGHOUSE ELECTRIC CORPORATION

1.0 INTRODUCTION

In a submittal of July 28, 1995 (Ref. 1), the Westinghouse Owners Group (WOG) requested U.S. Nuclear Regulatory Commission (NRC) review and approval of topical report WCAP-14416-P, "Westinghouse Spent Fuel Rack Criticality Analysis Methodology," June 1995 (Ref. 2). The report presents the current Westinghouse methodology for calculating the effective multiplication factor, k_{eff}, of spent fuel storage racks in which no credit is taken for soluble boron except under accident conditions. The report also presents a new proposed procedure for crediting soluble boron in the spent fuel pool water when performing storage rack criticality analysis for Westinghouse fuel storage pools. A revision to the methodology was submitted on October 23, 1996 (Ref. 28), based on recommendations by the NRC Committee to Review Generic Requirements (CRGR).

General Design Criterion (GDC) 62 (Ref. 3) states that "criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." The NRC has established a 5-percent subcriticality margin (k_{eff} no greater than 0.95) to comply with GDC 62 (Ref 4). All of the applicable biases and uncertainties should be combined with k_{eff} to provide a one-sided, upper tolerance limit on k_{eff} such that the true value will be less than the calculated value with a 95-percent probability at a 95-percent confidence level (Ref. 5). The proposed new methodology would permit the use of spent fuel pool soluble boron to offset these uncertainties to maintain k_{eff} less than or equal to 0.95. However, the spent fuel rack k_{eff} calculation would remain less than 1.0 (subcritical) when flooded with unborated water with a 95-percent probability at a 95-percent confidence level.

2.0 SUMMARY OF THE TOPICAL REPORT

Section 1.0 of the report is an introduction, stating the purpose of the report and summarizing the individual sections. Section 2.0 explains the computer codes used in the evaluation of the spent fuel rack k_{eff} calculations and presents benchmark results. In Section 3.0, the assumptions used to model the spent fuel storage racks and the reactivity effects of biases and uncertainties are presented. Section 4.0 discusses reactivity equivalencing methods that credit fuel assembly burnup and integral fuel burnable absorbers (IFBA). Section 5.0 describes postulated accidents that are considered in the spent fuel rack criticality analysis. Section 6.0 of the report, in conjunction with the supplement (Ref. 28), defines how credit for spent fuel pool soluble boron will be applied in the reactivity calculations.

3.0 TECHNICAL EVALUATION

The Westinghouse spent fuel rack criticality analysis methodology presented in WCAP-14416-P, and modified by Reference 28, provides a detailed description of both the current methodology, which has been used for many years by Westinghouse to calculate the reactivity of spent fuel storage racks, and a proposed new methodology with which partial credit for soluble boron in the pool water would be taken. The review of the proposed new methodology, given in Section 3.7 below, focused on the approximations and assumptions used as well as on revised technical specifications and analysis of dilution events required when crediting boron. The following evaluation is based on the material presented in the topical report, supplementary information (Ref. 28), discussions with Westinghouse staff, and responses to our requests for additional information (Refs. 14 and 26).

3.1 Computer Code Methods and Benchmarking

Reactivity calculations for the spent fuel storage racks are performed with the KENO-Va (Ref. 6) three-dimensional Monte Carlo computer code. A 227 energy group cross section library is created by NITAWL-II (Ref. 7) and XSDRNPM-S (Ref. 8) from ENDF/B-V data (Ref. 9). This method has been used to analyze a set of 32 low-enriched, water-moderated, UO_2 critical experiments to establish a method bias and uncertainty (Refs. 10, 11, 12, 13). These experiments cover a range of enrichments varying from 2.35 weight percent (w/o) to 4.31 w/o U^{235} separated by various materials (B_4C, borated aluminum, stainless steel, water) at fuel rod spacings from 0 to 6.56 cm. These experiments simulate current PWR spent fuel storage racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and neutron absorber worth. In response to a staff question (Ref. 14), WOG stated that no significant biases or trends were observed as a function of lattice or fuel parameters, including enrichment. The staff concludes that the KENO-Va benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to spent fuel storage rack conditions similar to those currently in use containing fuel rod enrichments up to 5.0 w/o U^{235}.

To minimize the statistical uncertainty of the KENO-Va calculations, at least 100,000 neutron histories are accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-Va reactivity calculations. In addition, edits from the KENO-Va calculations provide a visual inspection of the overall convergence of the results.

A method bias of 0.0077 results from the comparison of KENO-Va calculations with the average measured experimental k_{eff}. The standard deviation of the bias value is 0.00136 Δk. The 95-percent probability/95-percent confidence level (95/95) one-sided tolerance limit factor for 32 values is 2.20 (Ref. 15). Thus, there is a 95-percent probability with a 95-percent confidence level that the uncertainty in reactivity due to the method is not greater than 0.0030 Δk (2.20 x 0.00136).

The PHOENIX-P (Ref. 16) transport theory computer code is used to determine reactivity changes due to possible variations (tolerances) in material characteristics and mechanical dimensions in the fuel assembly and spent fuel racks, changes in pool conditions such as temperature and soluble boron, and fuel burnup. PHOENIX-P is a depletable, two-dimensional, multigroup, discrete-ordinates transport theory code that uses a 42 energy group nuclear data library.

PHOENIX-P has been compared with critical experiments (Refs. 17, 18, 19, 20). The PHOENIX-P reactivity predictions agree very well with the critical experiments, showing no significant bias or trends as a function of lattice or fuel parameters. The range of lattice parameters and configurations in the critical experiments encompassed present fuel storage configurations as realistically as possible.

PHOENIX-P has also been compared with isotopic measurements of fuel discharged from Yankee Core 5 (Ref. 21). The PHOENIX-P predictions agree very well with measurements for all measured isotopes throughout the burnup range.

Based on the above, we conclude that the analysis methods described are acceptable and capable of predicting the reactivity of PWR spent fuel storage racks containing assemblies with maximum fuel rod enrichments of 5.0 w/o U^{235} with a high degree of confidence.

3.2 KENO-Va Reactivity Calculations

KENO-Va is used to establish a nominal reference reactivity, using fresh (unirradiated) fuel assemblies and nominal rack dimensions, that satisfies the 0.95 k_{eff} acceptance criterion. The following assumptions are used in the calculation:

(1) The nominal spent fuel rack storage cell dimensions are used.

(2) Fuel assembly parameters for all assembly types considered for storage in the spent fuel pool are evaluated. These parameters include number of fuel rods per assembly, fuel rod clad material, fuel rod clad outer diameter, fuel rod clad thickness, fuel pellet outer diameter, fuel pellet density, fuel pellet dishing factor, fuel rod pitch, control rod guide tube material, number of guide tubes, guide tube outer diameter, guide tube thickness, instrument tube material, number of instrument tubes, instrument tube outer diameter, and instrument tube thickness.

(3) The nominal fresh fuel enrichment loaded into each fuel pin is modeled. The pin locations within a fuel assembly with multiple enrichments are considered, if applicable. The maximum fuel rod enrichment loaded into the fuel rods is limited to 5.0 w/o U^{235}.

(4) The nominal values for theoretical density and dishing fraction of the fuel pellets are modeled.

(5) If axial blankets are modeled, the length and enrichment of the blanket fuel pellets are considered.

(6) No amount of U^{234} or U^{236} is modeled in the fuel pellet.

(7) No amount of material from spacer grids or spacer sleeves is modeled in the fuel assembly.

(8) No amount of burnable absorber poison material is modeled in the fuel assembly.

(9) No amount of fission product poison material is modeled in the fuel assembly.

(10) The moderator is pure water (no boron) at a temperature of 68°F and a density of 1.0 gm/cc.

(11) If credit is taken for any fixed neutron-absorbing poison material panels present (except Boraflex), they are modeled using the as-built or manufacturer-specified poison material loadings and dimensions. Because of the significant Boraflex deterioration observed in some spent fuel racks, additional conservative assumptions are required for racks containing Boraflex as neutron absorber. These assumptions are not part of this technical review but will be reviewed on a case-by-case basis.

(12) If all storage cells are not loaded with the same fuel assembly type and enrichment, the specific storage configuration will be modeled. Different types of configurations include checkerboard patterns, empty cell locations, specific pool configurations, and other layouts as defined.

Using these assumptions, the spent fuel rack k_{eff} is calculated with KENO-Va to show that k_{eff} is less than or equal to 0.95 with no credit for soluble boron. A temperature bias, which accounts for the normal operational temperature range of the spent fuel pool water, and the method bias, determined from the benchmarking calculations, are included. In addition, if neutron absorber panels are used, a reactivity bias is added to correct for the modeling assumption that individual B^{10} atoms are homogeneously distributed within the absorber material rather than clustered around each B_4C particle. The staff concludes that these assumptions tend to maximize the rack reactivity and are, therefore, appropriately conservative and acceptable.

3.3 PHOENIX-P Tolerance/Uncertainty Calculations

PHOENIX-P is used to calculate the reactivity effects of possible variations in material characteristics and mechanical/manufacturing dimensions. The following tolerances and uncertainties are considered:

(1) Enrichment tolerance of ±0.05 w/o U^{235} about the nominal fresh reference enrichments

(2) Variation of ±2.0% about the nominal reference UO_2 theoretical density

(3) Variation in fuel pellet dishing fraction from 0% to twice the nominal dishing

(4) Tolerance about the nominal reference storage cell inner diameter, center-to-center pitch, and material thickness

(5) Tolerances about the nominal width, length, and thickness of neutron absorber panels

(6) Tolerances about the nominal poison loading of the neutron absorbing panels, if the nominal poison loading assumed in the KENO-Va model is not the minimum manufacturer-specified loading

(7) Asymmetric positioning of fuel assemblies within the storage cells

The manufacturing tolerance uncertainties are based on the reactivity difference between nominal and maximum tolerance values and, therefore, meet the 95/95 probability/confidence level requirement. These uncertainties are combined statistically with the 95/95 calculation uncertainty on the KENO-Va nominal reference k_{eff} and the 95/95 methodology uncertainty (0.0030 Δk) in the benchmarking bias determined for the KENO-Va methodology. The methodology benchmarking bias of 0.0077 Δk, the water temperature bias, and the B^{10} self-shielding bias, if applicable, are included in the final k_{eff} summation before comparison against the 0.95 k_{eff} limit. The following formula is used to determine the 95/95 k_{eff} for the spent fuel storage racks:

$$k_{eff} = k_{nominal} + B_{method} + B_{temp} + B_{self} + B_{uncert}$$

where:

$k_{nominal}$ = nominal conditions KENO-Va k_{eff}

B_{method} = method bias determined from benchmark critical comparisons

B_{temp} = temperature bias

B_{self} = B^{10} self-shielding bias, if applicable

$$B_{uncert} = \sqrt{\sum (\text{tolerance}_i \ldots \text{or} \ldots \text{uncertainty}_i)^2}$$

The staff concludes that the final k_{eff} calculated using the above methodology will satisfy the NRC guidance that the fuel storage rack reactivity be less than or equal to 0.95 when fully flooded with unborated water, including all appropriate uncertainties at the 95/95 probability/confidence level (Refs. 4, 5). Therefore, the documented methodology is acceptable.

3.3 Fuel Assembly Burnup Credit

Reactivity equivalencing is used to allow storage of fuel assemblies with higher initial enrichments (up to 5.0 w/o U^{235}) than those found acceptable using the previously described methodology. This concept is predicated upon the reactivity decrease associated with fuel depletion. For burnup credit, a series of reactivity calculations are performed with PHOENIX-P to generate a set of initial enrichment versus fuel assembly discharge burnup ordered pairs that all yield an equivalent k_{eff} (no greater than 0.95) when fuel assemblies are stored in the spent fuel storage racks.

The CINDER computer code (Ref. 22) was used to determine the most reactive time after reactor shutdown of an irradiated fuel assembly. CINDER is a point-depletion code that has been widely used and accepted in the nuclear industry to determine fission product activities. The fission products were permitted to decay for 30 years after shutdown and the fuel reactivity was found to reach a maximum at approximately 100 hours. At this time, the major fission product poison, Xe^{135}, has nearly completely decayed away. Therefore, the most reactive time for an assembly after shutdown of the reactor can be conservatively approximated by removing the Xe^{135}.

An uncertainty associated with the depletion of the fuel assembly and the reactivities computed with PHOENIX-P is accounted for in determining the reactivity equivalence limits. This uncertainty is based on the PHOENIX-P comparisons to the measured isotopics from the Yankee Core 5 experiments and is used to account for any depletion history effects or calculational uncertainties not included in the depletion conditions that are used in PHOENIX-P. The staff concludes that this uncertainty, which increases linearly with burnup from 0 at 0 burnup to 0.02 Δk at an assembly average burnup of 60,000 MWD/MTU, is conservative and acceptable.

The effect of axial burnup distribution on fuel assembly reactivity has been evaluated by modeling depleted fuel in both two dimensions and three dimensions. These evaluations show that axial burnup effects can cause assembly reactivity to increase at burnup-enrichment combinations greater than 40,000 MWD/MTU and 4.0 w/o U^{235}. Westinghouse has stated that this effect will be accounted for as an additional bias if burnup credit limits reach these combinations.

An additional conservatism is that the depletion calculations do not take credit for effects, such as Pu^{241} decay and Am^{241} growth, that are known to substantially reduce reactivity during long-term storage. However, the staff does not consider this to be a requirement.

The staff concludes that adequate conservatism has been incorporated in the methodology used to determine burnup credit.

3.4 Integral Fuel Burnable Absorber (IFBA) Credit

Another reactivity equivalencing technique for storage of fuel enrichments greater than those allowed by the previous methodology is based on the reactivity decrease associated with the addition of integral fuel burnable absorbers (IFBA) to Westinghouse fuel. IFBAs consist of neutron-absorbing material applied as a nonremovable thin zirconium diboride (ZrB_2) coating on the outside of the UO_2 pellet. PHOENIX-P is used to generate a set of initial assembly enrichment versus number of IFBA rods per assembly ordered pairs that all yield the equivalent k_{eff} (no greater than 0.95) when fuel assemblies are stored in the spent fuel storage racks. The following assumptions are used for the IFBA rod assemblies in the PHOENIX-P calculations:

(1) The fuel assembly is modeled at its most reactive point in life. This includes any time in life when the IFBA has depleted and the fuel assembly becomes more reactive.

(2) The B^{10} loading for each IFBA rod, determined from Westinghouse IFBA design specifications for the given fuel assembly type, is the minimum standard loading offered by Westinghouse for that fuel assembly type.

(3) The IFBA B^{10} loading is reduced by 5 percent to account for manufacturing tolerances and by an amount which corresponds to the minimum absorber length offered for the given fuel assembly type (e.g., a 144-inch fuel length with a minimum absorber length of 108 inches would result in a 25 percent IFBA B^{10} loading).

A calculational uncertainty of approximately 10 percent is included in the development of the IFBA requirements by adding an additional number of IFBA rods to each data point. To demonstrate that reactivity margin exists in the IFBA credit limit to accommodate future changes in IFBA patterns, calculations are also performed with nonstandard IFBA patterns. If a future change is made to the standard IFBA pattern designs, the reactivity difference between the new patterns and the old patterns will be calculated in order to assess the impact on both core reactivity and spent fuel rack IFBA credit limits.

The staff concludes that adequate conservatism has been incorporated in the methodology for determining IFBA requirements and that assemblies that comply with the enrichment-IFBA requirement curve developed by this methodology will have a k_{eff} no greater than 0.95 when placed in the spent fuel pool storage racks.

3.5 Infinite Multiplication Factor

An alternative method for determining the acceptability of fuel storage in a specific spent fuel rack is based on a PHOENIX-P calculation of the infinite multiplication factor (k_∞) for a fuel assembly in the reactor core geometry as a reference point. The fuel assembly model is based on a unit assembly configuration (infinite in the lateral and axial dimensions) in reactor geometry and is modeled at its most reactive point in life and moderated by pure water (no boron) at a temperature of 68°F with a density of 1.0 g/cc. A 0.01 Δk reactivity bias is added to this reference k_∞ to account for calculational uncertainties. The spent fuel storage rack is then modeled with these assemblies to ensure that the storage rack reactivity will be no greater than 0.95.

The staff concludes that fuel assemblies that have a reference k_∞ less than or equal to the value calculated with the above assumptions and methodology will have a k_{eff} no greater than 0.95 when placed in the spent fuel pool storage racks.

3.6 Postulated Accidents

The criterion that k_{eff} be no greater than 0.95 exists even for postulated accidents. Two types of accidents that can occur in a spent fuel storage rack may cause a reactivity increase: (1) a fuel assembly misplacement and (2) a pool water temperature change. However, for any of these accidents, the double contingency principle (Ref. 23) can be applied. According to this principle, it is unnecessary to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these postulated accidents, the presence of soluble boron in the pool water can be assumed as a realistic initial condition since assuming its absence would be a second unlikely event. PHOENIX-P boron worth calculations are used to determine the amount of soluble boron required to offset the highest reactivity increase caused by any postulated accident and to maintain k_{eff} less than or equal to 0.95, which is also the staff's acceptance criterion for accident conditions.

3.7 Soluble Boron Credit Methodology

In the proposed methodology for performing spent fuel rack reactivity calculations with credit for soluble boron in the pool water, a 95/95 rack k_{eff} is first calculated which remains below 1.0 (subcritical) with no soluble boron credit. This k_{eff} calculation uses the same assumptions described in Section 3.2 above, including the assumption of no soluble boron in the pool water. As previously described, a temperature bias, a method bias, a B^{10} self-shielding bias, and the 95/95 uncertainties associated with the calculation uncertainty, the methodology uncertainty in the benchmarking bias, and the manufacturing tolerances are included in the k_{eff} calculation.

The final equation for determining the k_{eff} requirement is

$$k_{eff} = k_{nominal} + B_{temp} + B_{method} + B_{self} + B_{uncert} < 1.0$$

where:

$k_{nominal}$ = nominal condition KENO-Va k_{eff}

B_{temp} = temperature bias for normal operating range

B_{method} = method bias from benchmark critical comparisons

B_{self} = B^{10} self shielding bias

$$B_{uncert} = \sqrt{\sum(\text{tolerance}_i\ldots\text{or}\ldots\text{uncertainty}_i)^2}$$

To determine the amount of soluble boron required to maintain $k_{eff} \leq 0.95$, KENO-Va is used to establish a nominal reference k_{eff} and PHOENIX-P is used to evaluate the reactivity effects of possible variations in material characteristics and mechanical manufacturing dimensions. These calculations contain the same assumptions, biases, tolerances, and uncertainties previously described except for the assumption regarding the moderator soluble boron concentration. Borated water is assumed instead of pure water. The tolerance calculations are, therefore, performed assuming the presence of soluble boron. The final 95/95 k_{eff} calculation is determined as described in Section 3.2 above and must be less than or equal to 0.95 with allowances for biases, tolerances, and uncertainties including the presence of the determined concentration of soluble boron.

For enrichments higher than those assumed in the k_{eff} calculation, reactivity equivalencing methodologies are used to determine burnup or IFBA credit. However, the maximum fuel rod enrichment is limited to 5.0 w/o U^{235}. Soluble boron credit is used to offset the uncertainties associated with each of these equivalencing methodologies, as appropriate.

Postulated accidents are considered in the same manner as discussed in Section 3.6 except that the previously determined amount of soluble boron for the 95/95 k_{eff} calculation, plus the amount determined for the reactivity equivalencing calculation, if required, is assumed present. The results of PHOENIX-P calculations of the reactivity change due to the presence of soluble boron are used to determine the amount of soluble boron required to offset the maximum reactivity increase caused by postulated accident conditions.

The final soluble boron credit requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm)

$SBC_{95/95}$ = soluble boron credit required for 95/95 k_{eff} less than or equal to 0.95 (ppm)

SBC_{RE} = soluble boron credit required for reactivity equivalencing methodologies (ppm)

SBC_{PA} = soluble boron credit required for k_{eff} less than or equal to 0.95 under accident conditions (ppm)

Thus the total soluble boron credit requirement will maintain the spent fuel rack k_{eff} less than or equal to 0.95 with a 95-percent probability at a 95-percent confidence level.

The total soluble boron required to maintain k_{eff} less than or equal to 0.95 is normally well below the large amount of soluble boron which is typically in spent fuel pool water. Therefore, a significant margin to criticality would generally still exist. However, a boron dilution analysis will be performed for each plant requesting soluble boron credit to ensure that sufficient time is available to detect and mitigate the dilution before the 0.95 k_{eff} design basis is exceeded and submitted to the NRC for review (Ref. 29). The analysis should include an evaluation of the following plant-specific features:

1. Spent Fuel Pool and Related System Features
   a) dilution sources
   b) dilution flow rates
   c) boration sources
   d) instrumentation
   e) administrative procedures
   f) piping
   g) loss of offsite power impact

2. Boron Dilution Initiating Events (including operator error)

3. Boron Dilution Times and Volumes

4.0 SUMMARY AND CONCLUSIONS

The topical report WCAP-14416-P and supporting documentation provided in References 14, 26 and 28 have been reviewed in detail. A major portion of this review focused on a proposed new methodology whereby partial credit could be taken for soluble boron in the spent fuel pool to meet the NRC-recommended criterion that the spent fuel rack multiplication factor (k_{eff}) be less than or equal to 0.95, at a 95-percent probability, 95-percent confidence level.

The staff concludes that the proposed new methodology for soluble boron credit is acceptable for the following reasons:

(1) Uncertainties in mechanical tolerances and storage rack dimensions are determined at the 95-percent probability, 95-percent confidence level and are incorporated in a conservative direction.

(2) Conservative uncertainties are incorporated for depletion calculations.

(3) A substantial margin to criticality would be available since the spent fuel rack k_{eff} will be less than or equal to 0.95, at a 95-percent probability, 95-percent confidence level, with an amount of soluble boron significantly less than that amount normally available in the pool.

(4) The fuel rack k_{eff}, will remain less than 1.0 (subcritical), at a 95-percent probability, 95-percent confidence level, even with no soluble boron in the spent fuel pool, thereby conforming to Criterion 62, "Prevention of criticality in fuel storage and handling" of Appendix A to 10 CFR Part 50.

The staff concludes that the methodology documented in WCAP-14416-P and Reference 28 can be used in licensing actions with the following provisions which are stated in WCAP-14416-P and Reference 28:

(1) If axial and planar variations of fuel assembly characteristics are present, they should be explicitly addressed, including the locations of burnable absorber rods.

(2) The maximum fuel rod enrichment shall be limited to 5.0 w/o U^{235}.

(3) The spent fuel storage racks should be assumed to be infinite in lateral extent or surrounded by a water reflector and concrete or structural material as appropriate to the design. The fuel may be assumed to be infinite in the axial dimension, or the effect of reflector on the top and bottom of the fuel may be evaluated.

(4) If credit for the reactivity depletion due to fuel burnup is taken, operating procedures should include provision for independent confirmation of the fuel burnup, either administratively or experimentally, before the fuel is placed in burnup-dependent storage cells.

(5) A reactivity uncertainty due to uncertainty in the fuel depletion history effects and depletion calculations should be included.

(6) A correction for the effect of the axial distribution in burnup should be determined and added to the reactivity calculated for uniform axial burnup distribution if it results in a positive reactivity effect.

In addition, as stated in the letter of October 18, 1996, from Westinghouse to the NRC (Ref. 28), the following items will be submitted by all licensees proposing to use the methodology described above:

(1) All licensees proposing to use the new method described above for soluble boron credit should submit a 10 CFR Part 50.36 technical specification change containing the following:

   a. k_{eff} shall be less than or equal to 0.95 if fully flooded with water borated to [1050] ppm which includes an allowance for uncertainties as described in WCAP-14416-P.

b. k_{eff} shall be less than 1.0 if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-P.

c. The spent fuel pool boron concentration shall be greater than [2300] ppm and shall be verified at a frequency of [7 days].

Licensees using the Westinghouse Improved Standard Technical Specifications (ISTS) described in NUREG-1431 (Ref. 27), should adopt specification 3.7.16, "Fuel Storage Boron Concentration," and 4.3.1, Fuel Storage-Criticality," as shown in section 5.0 below.

(2) All licensees proposing to use the new method described above for soluble boron credit should identify potential events which could dilute the spent fuel pool soluble boron to the concentration required to maintain the 0.95 k_{eff} limit (as defined in (1)a above) and should quantify the time span of these dilution events to show that sufficient time is available to enable adequate detection and suppression of any dilution event. The effects of incomplete boron mixing, such as boron stratification, should be considered. This analysis should be submitted for NRC review and should also be used to justify the surveillance interval used for verification of the technical specification minimum pool boron concentration.

(3) Although Boraflex deterioration is not addressed in this topical report, appropriate analyses are required to account for Boraflex degradation in storage racks that credit the negative reactivity effect of Boraflex. These analyses should be submitted for NRC review.

(4) Plant procedures should be upgraded, as necessary, to control pool boron concentration and water inventory during both normal and accident conditions.

5.0 TECHNICAL SPECIFICATIONS

3.7 PLANT SYSTEMS

3.7.16 Fuel Storage Pool Boron Concentration

LCO 3.7.16 The fuel storage pool boron concentration shall be ≥ [2300] ppm.

APPLICABILITY: When fuel assemblies are stored in the fuel storage pool and a fuel storage pool verification has not been performed since the last movement of fuel assemblies in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Fuel storage pool boron concentration not within limit.	------------NOTE------------ LCO 3.0.3 is not applicable. ----------------------------		
	A.1	Suspend movement of fuel assemblies in the fuel storage pool.	Immediately
	AND		
	A.2.1	Initiate action to restore fuel storage pool boron concentration to within limit.	Immediately
	OR		
	A.2.2	Verify by administrative means [Region 2] fuel storage pool verification has been performed since the last movement of fuel assemblies in the fuel storage pool.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.16.1	Verify the fuel storage pool boron concentration is within limit.	[7 days]

4.0 DESIGN FEATURES

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of [4.5] weight percent;

X b. $k_{eff} < 1.0$ if fully flooded with unborated water which includes an allowance for uncertainties as described in WCAP-14416-P;

X c. $k_{eff} \leq 0.95$ if fully flooded with water borated to [1050] ppm which includes an allowance for uncertainties as described in WCAP-14416-P;

[d. A nominal [9.15] inch center to center distance between fuel assemblies placed in [the high density fuel storage racks];]

[e. A nominal [10.95] inch center to center distance between fuel assemblies placed in [low density fuel storage racks];]

[f. New or partially spent fuel assemblies with a discharge burnup in the "acceptable range" of Figure [3.7.17-1] may be allowed unrestricted storage in [either] fuel storage rack(s); and]

[g. New or partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure [3.7.17-1] will be stored in compliance with the NRC approved [specific document containing the analytical methods, title, date, or specific configuration or figure].]

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CERTIFICATE OF DEPONENT

I, LAURENCE I. KOPP, do hereby certify that I have read the foregoing transcript of my deposition testimony and, with the exception of additions and corrections, if any, hereto, find it to be a true and accurate transcription thereof.

Laurence I. Kopp

12/27/99
DATE

Sworn and subscribed to before me, this the _____ day of _________, 19 _____.

NOTARY PUBLIC IN AND FOR

My commission expires:

ERRATA SHEET

PLEASE ATTACH TO THE DEPOSITION.

IN THE CASE OF : Carolina Power & Light
CASE # : 99-7162-02 LA

Please read the transcript of your deposition and make note of any errors in the transcription on this page. Do NOT mark on the transcript itself. Please sign and date the transcript on PAGE ______. Please return both Errata Sheet and transcript to : ______

PAGE	LINE	ERROR/AMENDMENT	REASON FOR CHANGE
5	1	change "C." to "I."	wrong middle initial
7	15	change "1993" to "1983"	typo
8	17	change "cancel" to "canal"	"
11	11	change "simply" to "simplify"	"
22	3	change "additional" to "official"	"
22	10	change "conparciances" to "concurrences"	"
29	6	change "wasn't" to "was"	"
29	12	change "200" to "with 2000"	"
29	15	change "puel" to "fuel"	"
38	6	change "are" to "effective"	"
38	9	change "solution" to "dilution"	"
39	7	change "plans" to "plants"	"
39	15	change "plan" to "plant"	"

ROBERT DEMPSTER & ASSOCIATES

OHN J. KEARNEY, Senior Vice President

DOCKETED USNRC

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February 17, 1984

Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Att'n: Docketing and Service Branch

Re: 10 C.F.R. Parts 2 and 72: Proposed Rule on Hybrid Hearing Procedures for Expansions of Onsite Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors

Gentlemen:

The following comments are submitted on behalf of the Edison Electric Institute (EEI) and the Utility Nuclear Waste Management Group (UNWMG). EEI is an association of the nation's investor-owned utilities; its members generate about seventy-eight percent of the nation's electricity and serve over sixty-seven million customers. UNWMG is comprised of forty-two utilities with specific interests relating to nuclear spent fuel storage. Its members are listed in Attachment A hereto. A significant number of the member utilities of EEI and UNWMG are likely to require expansion of onsite spent fuel storage prior to 1998 when the Department of Energy is committed to begin removal of spent fuel from the site of commercial nuclear power plants.

On December 5, 1983, the Commission published in the Federal Register a proposed rule that would amend its regulations at 10 C.F.R. Parts 2 and 72 to implement Section 134 of the Nuclear Waste Policy Act of 1982 (NWPA), which prescribed expedited licensing procedures for certain spent fuel storage technologies. 48 Fed. Reg. 54499 (1983). Consistent with the NWPA, the changes to existing Commission procedures would apply only to applications for a license or license amendment to expand onsite spent fuel storage capacity at commercial nuclear power reactors through the use of high-density fuel storage racks, fuel rod compaction, the transshipment of spent nuclear

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fuel to another civilian nuclear power reactor within the same utility system, the construction of additional spent nuclear fuel pool capacity or dry storage capacity, or by other means. The proposed new procedures would not apply to the first application for a license amendment to expand onsite fuel storage capacity by the use of a new technology not previously approved by the Commission for use at any nuclear power plant. Two options are identified in the proposed rule.

Option 1 substantially departs from the existing practice and would require the use of a "hybrid" hearing procedure in all proceedings to which Section 134 applies. For example, Option 1 removes the requirement that to be admitted to a licensing proceeding a petitioner must specify at least one valid "contention." Somewhat broader discovery would be allowed on any "issue" raised by intervenors and found to be within the scope of the proceeding. An "oral argument" procedure would be established as a means of determining those issues which should be adjudicated. Option 2 is a less drastic departure from existing rules and would provide a new summary disposition procedure utilizing oral argument, to be employed at the request of any party to the proceeding. As a result of the procedure in both proposed Option 1 and Option 2, the presiding officer would designate an issue for adjudication if there is a genuine and substantial dispute of fact which can be resolved with sufficient accuracy only by the introduction of evidence in an adjudicatory hearing and if the decision of the Commission is likely to depend in whole or in part on the resolution of such a dispute.

EEI/UNWMG finds that Option 1 is inconsistent with both the language of Section 134 of the NWPA and the legislative history and intent of Congress in enacting Section 134. While Option 2 is technically consistent with the wording of Section 134, it does not go as far as Congress intended in establishing meaningful procedural reform to provide an expedited proceeding for the expansion of spent fuel storage capacity at existing civilian nuclear power reactors. In this letter we propose modifications to Option 2 and additional procedures that are consistent with the Congressional mandate of Section 134.

The Legislative Purpose of Section 134 of the NWPA

Nowhere in the "Supplementary Information" published with the proposed rule nor inherent within the proposed changes to existing Commission procedural requirements does the Commission come to grips with the Congressional intent in enacting Section 134 of the NWPA. Nowhere does the Commission state its purpose in proposing changes to existing procedures other than to implement Section 134. Yet the intent of Congress in adopting Section 134 was clear, and it is just as clear that the Commission's proposal fails to accomplish what Congress intended.

The legislative history of the NWPA actually spans a period of over five years and three Congresses. During this period the utilities lobbied vigorously for a comprehensive Federal program for away-from-reactor interim storage of spent fuel. In finally passing the NWPA, the Congress did not establish the comprehensive Federal program for interim storage that the utilities had sought. Instead, Congress found that:

> [T]he persons owning and operating civilian nuclear power reactors have the primary responsibility for providing interim storage of spent nuclear fuel from such reactors, by maximizing, to the extent practical, the effective use of existing storage facilities at the site of each civilian nuclear power reactor, and by adopting new onsite storage capacity in a timely manner where practical. . . .

Section 131(a)(1) of the NWPA, 42 U.S.C. § 10151 (1983). The Congress did establish a limited Federal interim storage program to ensure that utilities did not lose full core reserve capability at the site of a nuclear reactor if diligent pursuit of onsite alternatives failed to provide in a timely manner for needed onsite storage capacity. Section 135 of the NWPA, 42 U.S.C. § 10155 (1983). But while the Congress found that utilities had the primary responsibility for spent fuel storage onsite, it also found that:

> [T]he Federal Government has the responsibility to <u>encourage and expedite</u> the effective use of existing storage facilities and the addition of needed new storage capacity at

> the site of each civilian nuclear power reactor. . . .

Section 131(a)(2) of the NWPA, supra (emphasis added). To accomplish this Congressional finding, Section 134 was adopted. Simply put, Section 134 was a trade-off; the utilities failed to convince Congress of the need for a major Federal program for interim storage of spent fuel, but Congress instead provided for expedited licensing of onsite spent fuel storage technologies.

While there is no Conference Committee report to provide a definitive legislative history of the NWPA, statements of the floor managers of the bills in the House and Senate during congressional debates and Committee Reports from the two houses leave little doubt as to the intent of the Congress in finally enacting Section 134. In the Senate, the precursor to the NWPA was S. 1662, a consensus bill drafted by members of the Senate Committee on Energy and Natural Resources and the Senate Committee on Environment and Public Works, and introduced by Senator McClure. During Senate debate of S. 1662, Senator Simpson explained to his colleagues the relationship between the proposed Federal interim storage policy in Title III of S. 1662 and the proposed changes to NRC procedures for expanding onsite spent fuel storage, as follows:

> Title III of the compromise provision establishes a firm, and I believe, appropriate national policy for the interim storage of spent fuel. Under this policy, the utility operators of nuclear powerplants bear the primary responsibility for interim storage of spent fuel at the sites of their nuclear plants. This places a significant, but appropriate, burden on the utilities to do everything possible to assure sufficient onsite storage capacity through a variety of measures specified in the legislation. These measures include reracking, transshipment of spent fuel between reactors within the same utility system, and the use of new technologies such as dry storage and the use of storage casks.

> In order to assist the utilities in carrying out this responsibility, the legislation contains measures to expedite the necessary regulatory approvals from the Nuclear Regulatory Commission for these various means of expanding onsite storage at reactor sites. These expedited licensing procedures in themselves represent an important step toward reforming this aspect of NRC's licensing process, while at the same time incorporating important safeguards to assure that the public health and safety is protected in these spent fuel storage expansion efforts.

128 Cong. Rec. S4157 (daily ed. April 28, 1982).

Senator Thurmond opposed Federal away-from-reactor spent fuel storage and offered an amendment to eliminate the provision for interim storage in S. 1662. In offering his amendment, Senator Thurmond reminded his fellow Senators that the "streamlined regulatory" process would remain:

> It should be stressed, however, that our amendment leaves intact those provisions of title III which establish a streamlined regulatory process to aid utilities in licensing additional storage space at reactor sites or in licensing transshipments of spent fuel to other sites.

128 Cong. Rec. S4274 (daily ed. April 29, 1982). Senator Simpson opposed the amendment and characterized the interim storage program in Title III as "'last resort,' emergency relief." Id. at S4281. Senator Simpson continued:

> This [national] policy [for spent fuel storage] places primary responsibility with the utilities for providing adequate spent fuel storage capacity at the reactor sites.
>
> [I]n order to carry out this element of the national policy, the bill includes new licensing procedures that are intended to expedite NRC approvals of utility requests for spent fuel storage expansion at reactor

> sites. These new procedures, which involve interim licensing authority and the use of hybrid hearing procedures, should minimize the potential for unnecessary delays in processing these utility license applications.

The Senate rejected Senator Thurmond's amendment (id. at :87) and subsequently passed S. 1662.

Section 313 of S. 1662 was similar to Section 134 of : subsequently enacted NWPA, although it did not limit issues it could be considered and did not proscribe the hybrid proiure in proceedings involving a new technology. During hearis on S. 1662, Chairman Palladino testified:

> S. 1662 has a number of important provisions with which we agree. It recognizes the need for additional storage facilities for spent fuel both onsite at reactors and at separate sites away from reactors; and the need to expedite the licensing activities related to expanding spent fuel capacity onsite at a reactor.

clear Waste Disposal: Joint Hearings on S. 637 and S. 1662 fore the Senate Comm. on Energy and Natural Resources and the ocomm. on Nuclear Regulation of the Senate Comm. on Environit and Public Works, 97th Cong., 1st Sess 236 (1981) (stateit of Nunzio J. Palladino, Chairman, NRC).

The efforts by the House of Representatives to pass a mprehensive nuclear waste bill during the 97th Congress were re complicated. Three major committees -- Interior and Insur Affairs, Science and Technology, and Energy and Commerce -- d jurisdiction, and each reported and approved separate lls: H.R. 3809, reported in H.R. Rep. No. 491, Part 1, 97th ng., 2d Sess. (1982); H.R. 5016, reported in H.R. Rep. No. .1, Part 1, 97th Cong., 1st Sess. (1981); and H.R. 6598, ported in H.R. Rep. No. 785, Part I, 97th Cong., 2d Sess. 982). Subsequently, the three committees entered into negoations to reconcile H.R. 3809, H.R. 5016, and H.R. 6598. The sult of the negotiations was H.R. 7187. This bill was presited to the House on September 30, 1982 as a substitute amendint to H.R. 3809. 128 Cong. Rec. H8162 (daily ed. September

30, 1982). It was passed by the House on December 2, 1982. 128 Cong. Rec. H8800 (daily ed. December 2, 1982).

Many of the provisions of H.R. 7187, including Section 134, were drawn from H.R. 6598 as amended by the Committee on Energy and Commerce.1/ Section 134 had been added to H.R. 6598 by the Committee on Energy and Commerce; it provided for hybrid hearings in license or license amendment proceedings to expand onsite spent nuclear fuel storage capacity, restricted the issues that could be litigated in such a proceeding, and authorized interim licensing. The Committee explained:

> Procedural changes are made to the NRC licensing process to encourage utilities to expand storage capacity at reactor sites. Except for the use of a technology which has been adopted on a generic basis, each of the methods for expanding storage capacity requires a license or an amendment to the existing operating license. The bill provides for expediting the consideration of such application by "scoping" issues in an informal oral argument preceded by discovery and requiring at the conclusion of such informal oral argument that the Commission designate disputed questions of fact and law for formal adjudication only if it determines there is a genuine disputed issue of fact and the Commission's decision is likely to depend in whole or in part on the resolution of the issue(s) they seek to raise in order to be granted an adjudicatory hearing. In any Commission proceeding to expand spent fuel storage capacity, six categories of issues, such as need for power generated by the reactor involved, would be excluded from consideration. In addition the Commission is authorized to grant an interim license or interim amendment to an existing licence for expansion of onsite storage or transshipment prior

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1/ 128 Cong. Rec. H8168 (daily ed. September 30, 1982) (statement of Rep. Dingell).

[to] the conduct or completion to any hearing required by law, provided that in all other respects the requirements of the law are met and there is assurance that public health and safety will be protected and refusal to grant an interim license would prevent a petitioner from providing adequate onsite storage capacity.

.. Rep. No. 785, Part I, 97th Cong., 2d Sess. 39 (1982).2/

H.R. 6598 as amended also authorized a limited amount Federal interim spent fuel storage; but the Committee made clear that onsite capacity was the primary means of interim rage. Similar to section 301 of S.1662 (the bill which the ate had passed), Section 131 of H.R. 6598 established the icy of "maximizing, to the extent practical, the effective of existing storage facilities at the site of each civilian lear power plant." The Committee added,

> The Federal Government is charged with the responsibility to provide limited "last resort" interim storage capacity for civilian nuclear power reactors determined by the Nuclear Regulatory Commission to be needed to assure the continued orderly operation of the reactor, through the maintenance of full core reserve storage capability.

. Rep. 785, Part 1, 97th Cong., 2d Sess. 39 (1982).

H.R. 7187 eliminated the interim licensing authority at had been included in Section 134 of H.R. 6598. H.R. 7187 so reformulated the issues that were excluded from the scope hybrid proceedings. Except for the absence of subsection)(4) on the licensing of new technology, which was added by sequent amendment, Section 134 of H.R. 7187 was identical to subsequently enacted provision.

Representative Dingell suggested that H.R. Rep. No. 785 be nsidered part of the legislative history of H.R. 7187. 128 ng. Rec. H8168 (daily ed. September 30, 1982).

During the floor debates, the hybrid provisions were mentioned only incidentally, during discussions on an amendment to eliminate the provisions for federal interim storage; but these comments again emphasized the intent of Congress to expedite expansion of onsite storage. Rep. Lundine, the proponent of the amendment, stated

> My amendment would also preserve provisions in the bill for expedited NRC licensing provisions for at reactor interim storage. These streamlined procedures at the NRC will insure timely action on licensing issues.

128 Cong. Rec. H8581 (daily ed. November 30, 1982). Similarly, Rep. Broyhill, in opposing Rep. Lundine's amendment, stated

> Well, another purpose of the bill is this: Section 131 . . . continuing through sections 132, 133, and 134, provides for expedited consideration of applications for expansion of onsite storage of these spent fuels, and certainly there is a crying need for these expedited procedures. Generally speaking, I would say there is agreement that these expedited procedures for the licensing of these onsite facilities are needed, and if there is no final resting place by 1998, obviously there is going to have to be some consideration for the expansion of onsite storage.

Id. at H8584. Rep. Lundine's amendment was subsequently rejected. Id. at H8590.

The Senate and House bills that had been passed -- S. 1662 and the text of H.R. 7187 as H.R. 3809 -- were not referred to a House-Senate conference; instead, in order to expedite the legislation, the Senate Committee on Energy and Natural Resources introduced an amended version of the House-passed bill. 128 Cong. Rec. S15621, S15639-42, S15669 (daily ed. December 20, 1982). On December 20, 1982, the Senate passed this bill, and the House then agreed to the Senate amendments. Id. at S15670, H10525.

The Senate bill amended Section 134, but only to add a restriction on the use of the hybrid procedure; the hybrid procedure was prohibited in proceedings to expand onsite fuel storage by the use of new technology. Id. at S15643-44. However, debates on this amendment once again stressed that the purpose of Section 134 was to expedite:

> Section 134 of the McClure substitute amendment to H.R. 3809 provides for an abbreviated, legislative-type hearing to procede the normal full adjudicatory hearing. The purpose of the abbreviated hearing is to speed up the licensing of onsite storage expansion. A full hearing would only be necessary if it were determined that a "genuine and substantial dispute of fact" exists; that such dispute could be resolved in a full adjudicatory hearing; and that the decision of the Commission is likely to depend in whole or in part on the resolution of the dispute. The criteria by which the Commission may decide that a full adjudicatory hearing is necessary is extremely narrow.

Id. at S15644 (statement of Sen. Mitchell).

Thus it is indisputable that the intent of Congress in enacting Section 134 of the NWPA was to expedite the licensing of expanded onsite storage. Congress perceived an expedited licensing procedure as essential if powerplant operators were to bear the burden of supplying sufficient interim storage capacity. As we discuss below, neither Option 1 nor Option 2, as proposed by the Commission addresses this clear legislative intent.

OPTION I

EEI/UNWMG strongly opposes Option 1. The procedure set forth in Option 1 is inconsistent with the NWPA. For example, Option 1 is not optional and therefore does not comply with the statutory mandate that an opportunity for oral argument be provided "at the request of any party." 42 U.S.C. § 10154(a) (1983). Moreover, not only does the procedure proposed in Option 1 fail to satisfy the clear legislative intent

--to expedite the licensing process and hence the expansion of onsite spent fuel storage capacity -- but in our view it is likely to lengthen the process.

We provide a section-by-section analysis of Option 1 in Attachment B hereto. Briefly, we have the following specific concerns with respect to Option 1. It eliminates contention pleading requirements. Thus, discovery is wide-open to any "issue" which an intervenor may wish to raise. Indeed, the Commission admits "discovery will be somewhat broader than under existing practice." 48 Fed. Reg. at 54501. Also, Option 1 eliminates the traditional "one good contention" rule for party status. The implied corollary is that an intervenor remains a party to a proceeding even if all its allegations are summarily resolved. Finally, Option 1 permits cross-examination during the oral argument and calls for formal findings and conclusions. This procedure far exceeds "oral argument" and borders on formal adjudication; and it implies that all issues which are not designated for adjudication must be decided by the Licensing Board, whereas dismissal might be appropriate.

Presumably the benefit the Commission believes would result from the hybrid procedure proposed in Option 1 is the disposition of most if not all issues after oral argument, thus avoiding or narrowing the scope of a hearing. Yet, as noted by the Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals on the Proposed Nuclear Licensing Reform Act of 1983:

> [I]n general, the major delays associated with public hearings are attributable to the time devoted to getting to the public hearing and to the time required to obtain decisions following the public hearing. With rare exceptions, the public hearings themselves -- even with protracted cross examination -- have not been a material schedule factor in the overall public hearing process.

Report of the Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals on the Proposed Nuclear Licensing Reform Act of 1983 (December 15, 1982), at 14. We submit that Option 1 would allow an intervenor intent on delay ample opportunity to bog down the process with unbridled discovery, an untried and unduly formal oral argument procedure, and the possibility of a hearing in any event on some issues after this process. Indeed, often licensees, when the schedule for obtaining a license amendment is crucial, avoid filing motions for summary disposition of issues in a proceeding -- even where the result almost surely would be favorable -- in order to avoid the delay in getting to hearing and to a decision. The unanimous judgment of attorneys representing utility members of EEI/UNWMG is that Option 1 would lengthen the process, and would thus utterly fail in achieving Congressional purpose.

OPTION 2

Option 2 is much preferable to Option 1. The normal rules for the pleading and admissibility of contentions apply, and this procedure ensures that only specific, controverted matters are referred to the hybrid procedure.3/ In addition, the one good contention rule remains in effect and allows the dismissal of intervenors who fail to advance litigable issues. Also, Option 2 makes the hybrid procedure optional, consistent with the Act, and conforms more closely to the procedures prescribed by the Act. In particular, Option 2 does not authorize cross-examination during oral argument.

On the other hand, not all of the problems in Option 1 are eliminated in Option 2. The Commission ignored the invitation from Congress to fashion expeditious rules of discovery particularly applicable to this type of proceeding. Option 2 places no time limits on discovery, and specific limits are absolutely essential if the hybrid procedure is to expedite the licensing process. See § 2.749a(b). Also, like Option 1, Option 2 does not provide for prefiled sworn testimony and written submissions, and the prefiling of this material would permit the parties to better prepare for oral argument. Id. Similarly, Option 2 does not make it clear that the "written submissions" which may be relied upon refer to sworn written or documentary material admissible as evidence. Id. In addition, like Option 1, Option 2 provides that the presiding officer

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3/ To the extent there may be some ambiguity in the wording of Option 2, we have proposed clarifying language which makes it clear that the oral argument procedure is available only for disposition of contentions previously admitted.

shall only consider those facts and data submitted in the form of sworn testimony or written submission. Id. This provision may inappropriately preclude official notice and is inconsistent with the Act. See discussion of Option 1, § 2.1105(b) at Attachment B. Finally, Option 2 calls for a decision supported by formal findings and conclusions on issues not designated for adjudication.

Although EEI/UNWMG finds Option 2 preferable to Option 1, it cannot support Option 2 in its present form. Option 2 merely replaces one summary disposition procedure with another (albeit with an improved standard) and therefore does relatively little to expedite the licensing process. For that reason, it ignores the clear intent of the NWPA and squanders the opportunity to develop an innovative and efficient hearing process for licensing spent fuel storage technologies. Accordingly, EEI/UNWMG strongly recommends adoption of the additional procedures discussed below. Attachment C hereto sets forth the actual text of our proposed modifications and additional provisions to be incorporated with Option 2.4/

Additional Changes to the Commission's Rules to Implement Congressional Intent in Enacting Section 134 of the NWPA

To remedy the problems discussed supra with respect to Option 2, EEI/UNWMG proposes certain changes to the proposed Section 2.749a. These changes include: 1) amending subsection (b) to clarify the evidentiary nature of "written submissions;" 2) amending subsection (b) to require the prefiling of sworn testimony and written submissions; 3) amending subsection (b)

4/ While not part of the proposed rule, EEI/UNWMG strongly endorses the Commission's strict interpretation of the "Sholly Amendment" as it applies to applications for expansions of onsite spent fuel storage technologies. See 48 Fed. Reg. at 54500, note 1. We anticipate that license amendments to permit spent fuel storage expansions generally will not involve a "significant hazard" consideration and such license amendments can be issued immediately. Thus, the expedited procedures that we proposes here are in all parties' interests, particularly where the hearing process is available only subsequent to the issuance of the license amendment.

so that official notice is not inadvertently precluded; 4) amending subsection (c) so that the presiding officer is authorized to dismiss issues; and 5) amending subsection (c) and deleting subsection (f) to eliminate formal findings. See Attachment C at 3-5.5/

EEI/UNWMG also believes that the Commission must address the question of procedural reform, consistent with Section 134, that will meaningfully expedite the licensing process.6/ In this regard, in addition to the modifications to Option 2 proposed above, we propose that Commission

(1) adopt a threshold *prima facie* test for admission of a contention;

(2) limit discovery to the scope of admitted contentions and no more than two rounds, to be completed during a period established by the presiding officer not to exceed ninety days; and

(3) establish by rule criteria to be considered by a Board, after hearing oral argument pursuant to the Option 2 procedures, in determining whether a contention should be litigated in an adjudicatory proceeding.

A. Contentions

The present rules governing admissibility of contentions, which require the party offering a contention simply to state the basis for the contention with reasonable specificity, is inappropriate for proceedings involving expansions of onsite

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5/ A number of clarifying changes have also been proposed to Section 2.749a. For example, EEI/UNWMG proposes changing the words "matters" and "issues" to "contentions," since Section 2.749a applies to admitted contentions.

6/ Such procedural reforms as we propose here may or may not be appropriate for consideration in the broader context of overall licensing reforms. That issue is not addressed here.

spent fuel storage capacity. Atomic Safety and Licensing Boards have noted that existing regulations require admission of a properly pleaded contention even where the same generic issue has been previously litigated in other proceedings and found wanting.7/ Once a contention is admitted, the intervenor can require the applicant to invest considerable time and expense in the discovery process -- answering interrogatories and sorting through often massive numbers of documents. Based on the collective experience of the members of EEI/UNWMG, we believe no single change to the regulations is more likely to expedite consideration of licensing issues than to require an intervenor at the outset of the hearing process to establish a prima facie showing that he has information available to support his allegation sufficient "to require reasonable minds to inquire further." Such information may take the form of affidavits, technical reports or articles in technical and scientific publications. Indeed, the Commission itself has proposed as one alternative means of expediting all licensing proceedings that the prima facie showing be generally required before contentions could be admitted. Notice of Proposed Rule - Modifications to the NRC Hearing Process, 46 Fed. Reg. 30349, 30350-1 (1981). That proposal is still pending. See NRC Regulatory Agenda, 48 Fed. Reg. 48156, 48160 (1983). Note also that a petitioner preparing contentions in a hybrid spent fuel proceeding will not only have available to him the information contained in the application for a license amendment, but will also be able to refer to the record of the proceeding in which

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7/ The Commission's Rules an admission of contentions have been interpreted such that even frivolous issues may not be rejected on the obvious merits of the issue but must wait summary disposition or adjudication. See Mississippi Power & Light Co. (Grand Gulf Nuclear Station, Units 1 and 2), ALAB-130, 6 A.E.C. 423, 426 (1973) (admitting an alternative source contention despite the fact that geothermal sites, on which the contention was based, did not exist in applicant's service area); see also Carolina Power & Light Co. (Shearon Harris Nuclear Power Plant, Units 1 and 2), LBP-82-119A, 16 N.R.C. 2069, 2103 (1982) (in which the Board admitted part of a contention which postulated the unlikely fouling of the condensers by clams; oysters, or barnacles; there the Board stated: "Had we any authority to reject a contention on its merits, we would reject this clam and barnacle scenario because we can scarcely imagine that it would present a safety problem, as alleged.")

the pertinent technology was first licensed. For a number of the spent fuel storage technologies being developed in joint utility-DOE demonstration programs, the technical reports of those demonstration programs wil also be available.

The intent in proposing this higher threshold for admission of contentions is to eliminate frivolous contentions, to eliminate repetitive consideration of generic issues in each spent fuel storage expansion licensing proceeding, and to eliminate scatter-shot pleading of contentions in quantity with little attention to quality. It would allow Boards to make certain discretionary technical judgments on the likely ability of an intervenor to raise an issue of technical substance. It would not preclude an intervenor from raising issues which have any real technical merit.

B. Discovery

The most time consuming aspect of the hearing process is discovery. While the Congress clearly contemplated that the oral argument mandated by Section 134 would be preceded by discovery, it did not provide that the Commission would simply adhere to present discovery rules. Indeed, Section 134 provides that the oral argument "shall be preceded by such discovery procedures as the rules of the Commission shall provide." (emphasis supplied). The use of the future tense invites the Commission to examine the role of discovery in an expedited hybrid procedure. EEI/UNWMG proposes that the Commission modify the discovery procedures as follows:

(1) each party would be limited to two rounds of discovery on admitted contentions (whether by deposition, interrogatory, production of documents, or a combination thereof); the second round would be limited to follow-up questions regarding responses to the first round;

(2) all discovery would be conducted during a period established by the presiding officer, not to exceed ninety days;

(3) the time limitations established for responding to discovery requests would be strictly enforced absent consent of the parties and good cause shown; and

(4) the Board would rule on discovery disputes expeditiously, consistent with the period established for conducting all discovery.

These modest modifications to the existing rules for discovery are similar to procedures which have been adopted by Boards in licensing proceedings under present rules. They provide for greater discipline and focus in the discovery phase of a hearing. These modifications would balance the intent to expedite the hearing procedure with the objective of discovery to remove the element of surprise in modern administrative practice.

C. Additional Criteria to be Applied by the Board in Determining Whether an Issue Heard in Oral Argument Should be Litigated in an Adjudicatory Hearing

In its discussion of the proposed rule, the Commission noted:

> The criteria that the presiding officer must apply in determining which issues, if any, should be resolved in an adjudicatory hearing are identical to the statutory language. The standard is quite strict and is intended to ensure that the resources of all parties to any adjudicatory hearing are focused exclusively on real issues.

48 Fed. Reg. at 54501. EEI/UNWMG submits that the Commission should provide additional criteria to assist Boards in determining whether there is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing and whether the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute. In this regard we suggest the following criteria:

(1) Where an issue has previously been considered in another proceeding regarding the licensing of the same technology for spent fuel storage or transshipment, the party sponsoring the contention must

> demonstrate the existence of significant new or differing information that is likely to render the earlier findings of the Commission incorrect.
>
> (2) A party must make a showing that its contentions will be supported by sworn testimony or exhibits sponsored by a qualified expert.
>
> (3) Where a contention involves only differing technical judgments applied to an undisputed set of facts, the Atomic Safety and Licensing Board may determine that adjudication is not necessary where NRC Staff Regulatory Guides or other credible published technical information establish a clear consensus of the scientific community regarding the issue raised by such a contention.

(1) It was the intent of Congress to encourage the utilization of demonstrated spent fuel storage technologies at more than one site. Indeed, Congress established a joint DOE-industry program to encourage the early demonstration of onsite spent fuel storage technologies and the licensability of such technologies. See Section 218 of the NWPA, 42 U.S.C. § 10198 (1983). Where the application of any such technology has already been licensed it is reasonable to establish a presumption of technical acceptability that an intervenor should be required to rebut before allowing a technical issue to be litigated at subsequent proceedings. Compare 10 C.F.R. §2.503.

(2) Where a technology has previously been licensed at the site of another nuclear plant, it is extremely unlikely that an intervenor could ever establish a technical case against that technology absent the introduction of compelling factual evidence. Thus, one of the criteria for designating a contention for adjudication should be a showing by the intervenor that he is prepared to introduce facts sponsored by a qualified expert to support his contention. Absent such a showing, it is unlikely that the intervenor could make a meaningful contribution to the record on such a factual issue.

(3) Often issues in licensing proceedings do not in-lve "a genuine and substantial dispute of fact" so much as a ffering technical judgment applied to an undisputed set of cts. Just as often these differing technical judgments are ther unsupported or are advanced by witnesses whose views are jected by the vast majority of the relevant scientific commu-ty. Consistent with the Congressional mandate to adjudicate ly those issues where "there is a genuine and substantial spute of fact which can only be resolved with sufficient ac-racy by the introduction of evidence in an adjudicatory hear-g," the Atomic Safety and Licensing Boards should reject con-ntions that do no more than advance such discredited chnical judgments, especially where the consensus of the sci-tific community has been published in an ANSI standard, NRC gulatory Guide, or other credible technical reports.

Conclusion

EEI/UNWMG believes that the Commission must focus once ;ain on Section 134 of the NWPA and its clear legislative in-nt. As we took some pains to demonstrate, that intent was to dite the licensing process for a relatively narrow subset license proceedings. The next question we addressed is iether the Commission's proposed rule would accomplish what ongress intended. With respect to Option 1, the answer is em-iatically no -- it most likely would achieve the opposite re-ilt. While Option 2 is generally acceptable and offers some odest improvement in the procedures for summary disposition, t also fails to meet Congress' intent. However, by adopting ne modifications to Option 2 and the additional procedures roposed in this letter, the Commission would accomplish mean-ngful reform to its licensing procedures -- for this special ubset of license proceedings -- and would implement the letter nd spirit of the Congressional mandate of Section 134.

Respectfully submitted,

John J. Kearney

John J. Kearney
Senior Vice President

ttachments

Attacnment A

UNWMG

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MEMBERS

Alabama Power Company
Arizona Public Service Company
Baltimore Gas & Electric Company
Boston Edison Company
Carolina Power & Light Company
The Cincinnati Gas & Electric Company
The Cleveland Electric Illuminating Company
Commonwealth Edison Company
Consolidated Edison Co. of New York, Inc.
The Detroit Edison Company
Duke Power Company
Duquesne Light Company
Florida Power Corporation
Florida Power & Light Company
Georgia Power Company
Gulf States Utilities Company
Houston Lighting & Power Company
Illinois Power Company
Iowa Electric Light & Power Company
Long Island Lighting Company
Los Angeles. Department of Water & Power
Middle South Services, Inc.
Nebraska Public Power District
Niagara Mohawk Power Corporation
Northeast Utilities
Northern States Power Company
Pennsylvania Power & Light Company
Philadelphia Electric Company
Portland General Electric Company
Power Authority of the State of New York
Public Service Electric & Gas Company
Sacramento Municipal Utility District
SNUPPS
- Union Electric Company
- Kansas Gas & Electric Company
- Kansas City Power & Light Company

Southern California Edison Company
Texas Utilities Company
Toledo Edison Company
Virginia Electric & Power Company
Wisconsin Public Service Corporation
Wisconsin Electric Power Company
Yankee Atomic Electric Company

ATTACHMENT B

Section-by-Section Analysis of Option 1

1. § 2.1100 -- Purpose

This section omits any reference to the purpose of the hybrid procedures--to expedite the licensing process. Since some rules of procedure will inevitably be determined by Licensing and Appeal Boards implementing and interpreting the regulations, a clear and authoritative statement of purpose is essential. Indeed, the absence of such a statement might be interpreted as a repudiation of the intent to expedite.

2. § 2.1101--Scope of subpart.

Section 2.1101 requires the use of hybrid hearing procedures in all proceedings to which Section 134 of the Nuclear Waste Policy Act applies. Section 134, however, prescribes an optional procedure--a procedure to be used only "at the request of any party." 42 U.S.C. § 10154(a) (1983). Not only is Section 2.1101 in clear violation of the Act, but it also needlessly eliminates adjudicatory flexibility. There is no justification for the use of hybrid procedures if all parties to a particular proceeding find normal procedures advantageous.

3. § 2.1103--Requests for hearing or petitions to intervene.

This section abandons the "one good contention" rule. See 10 C.F.R. § 2.714(b). Instead, any petitioner will be made a party if he or she satisfies standing requirements. Such a procedure would open the floodgates for nuisance intervention. Petitioners with no real dispute would be granted party status, entitling them to the hybrid procedure, including discovery. An applicant could well be overwhelmed by the number of participants and their pleadings and requests. At best, this scenario would cause substantial delay; at worst, it would deter onsite storage expansion.

In addition, because one good contention would no longer be a prerequisite to party status, there would be no basis for dismissing a party who subsequently fails either a) to state an issue within the scope of the proceeding (see § 2.1104(a)), or b) to raise a genuine or substantial dispute of fact for which adjudication is necessary (see § 2.1106(b)). Consequently, parties initially admitted would remain so; and if each were permitted to cross-examine witnesses testifying on another party's contention, as is presently the practice, the eventual adjudicatory hearing would be unnecessarily lengthened.

4. § 2.1104--Filing of list of issues; requests for oral arguments.

This section abandons the traditional "basis with specificty" requirement for contentions. See 10 C.F.R. § 2.714(b). Instead, a party need only allege "issues." No criteria are provided to determine whether an "issue" is sufficient as a matter of pleading. In fact, a party might simply restate the "aspects of the subject matter" which it identified in its original petition to intervene. See § 2.1103(b). An issue could easily be phrased in such broad terms (and still be within the scope of the proceeding) that it would require several rounds of discovery merely to determine the true matter in controversy--a prerequisite to any meaningful discovery on the factual bases of such an issue. Again, such a procedure could create significant delays.

Section 2.1104's elimination of admitted contentions is perpetuated throughout the remainder of Option 1. Section 2.1105 repeats the explicit requirement of the NWPA that the hybrid procedure be limited "to those matters in controversy among the parties." Compare 42 U.S.C. § 10154(a) (1983). But because Option 1 excludes procedures for formulating and ruling on contentions, it provides no mechanism for determining "matters in controversy." As a result, Option 1 permits oral argument, preceded by discovery, of undefined (and possibly frivolous or nonsensical) "issues" that raise no real controversy;

hence, it clearly violates the "matter in controversy" limitation.

5. § 2.1105 -- Discovery; oral argument.

As discussed above, Option 1 does not restrict the use of the hybrid procedures set forth in Section 2.1105 to "matters in controversy" and therefore violates the NWPA. In addition, there are a number of other problems in this section.

Subsection (a) provides that discovery shall begin and end at such times as the presiding officer shall determine. However, it provides no guidance as to the appropriate length of discovery. In order to assure expeditious proceedings, specific time periods for discovery should be prescribed.

Subsection (b) provides for the submission of a summary of facts, data, and arguments fourteen days prior to oral argument. The subsection adds that "[o]nly facts and data in the form of sworn testimony or written submission may be relied upon by the parties during oral argument, and the presiding officer shall consider only those facts and data submitted in such form."

Subsection (b) is ambiguous in several respects. First, it should be made clear that "written submission" should be attested to or otherwise admissible as evidence. Otherwise, such material would not provide a reliable and rational basis for a finding that a substantial dispute of fact exists. Second, the subsection does not state when the sworn testimony or written submissions are to be submitted, although it implies that these materials are submitted at the oral argument. However, for there to be a meaningful oral argument, these materials should be prefiled, as is customary in NRC proceedings. Accordingly, the subsection should provide that sworn testimony and written submission accompany each party's summary. Finally, subsection (b) precludes official notice. The preclusion is not supported by the Act, which states "[o]f the materials that may be submitted, the Commission shall only consider those facts and data that are submitted into the form of sworn testimony or written submission." 42 U.S.C. § 10154 (1983) (emphasis added). As in any adjudication, the presiding officer should be able to use indisputable adjudicative facts and use legislative facts in rulings on law or policy.

Subsection (c) of Section 2.1105 permits cross-examination during the oral arguments, at the discretion of the presiding officer. The subsection, however, provides no criteria for determining when cross-examination is appropriate, and quite conceivably the "oral argument" might evolve into a full scale adjudication (sworn testimony and cross-examination) on relatively undefined issues. The resulting delay would be considerable. Accordingly, discretionary cross-examination should be eliminated. Cross-examination is inconsistent with the intent of Congress, which envisioned a "legislative-type hearing." It is inconsistent with "oral argument," could lead to two tiers of formal adjudication, and would result in substantial delay.1/

6. § 2.1106--Designation of issues for hearing

Section 2.1106 requires that after oral argument the presiding officer issue an order designating the facts to be adjudicated. The Section further provides that the order shall include "a statement of findings and conclusions, together with the reasons or basis for them, with respect to any issue heard at the oral argument that is not designated for resolution at the adjudicatory hearing." The Act, however, does not call for such formal findings and conclusions. The procedure is inconsistent with informal, legislative-type hearings. The presiding officer should not be required to "decide" all issues not designated for adjudication -- issues which the presiding officer may determine to be insubstantial or inappropriate for resolution by adjudication. Instead, the presiding officer should simply dismiss such issues,2/ and the presiding officer's

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1/ Note that if sworn testimony and written submission were not prefiled, as suggested above, the parties would be unprepared to conduct meaningful cross-examination.

2/ Dismissed issues might be referred to the NRC staff for informal, nonadjudicatory resolution.

determination should merely be supported by an adequate statement of reason. <u>See</u> 5 U.S.C. § 555(e) (1983). Otherwise, it may take months before the presiding officer issues his decision.3/

Subsection (e) of Section 2.1106 provides that the presiding officer's designation of issues is interlocutory, and appeals must await the end of the proceeding, except to the extent authorized by 10 C.F.R. § 2714a. 10 C.F.R. § 2.714a, however, only permits appeal on the question whether or not to wholly deny a petition to intervene; and under Option 1, this question is decided irrespective of the issues pleaded by petitioner. <u>See</u> § 2.1103. Accordingly, § 2.714a does not provide an exception to the rule proposed in § 2.1106(e) and reference to § 2.714a should be deleted.

3/ This potential for delay would be especially great if there were no mechanism for designating admissible, specific contentions prior to the invocation of hybrid procedures.

ATTACHMENT C

Proposed Revisions to Option 2 1/

[Section 2.4 (Definitions) is amended by adding:

(t) Spent fuel proceedings, pursuant to Part 50 this chapter, include an application for a license, for an amendment to an existing license, filed after nuary 7, 1983 and prior to December 31, 2005, to exnd the spent nuclear fuel storage capacity at the te of a civilian nuclear power reactor, through the e of high-density fuel storage racks, fuel rod comction, the transshipment of spent nuclear fuel to an- civilian nuclear power reactor within the same ility system, the construction of additional spent clear fuel capacity or dry storage capacity, or by her means.]

[Section 2.714(b) is amended by adding, after the first ntence, the following:

spent fuel proceedings other than on the first application r a license or license amendment to expand on-site fuel

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' Bracketed text indicates proposed additions. Overstruck xt indicates proposed deletions.

:orage capacity by the use of a new technology not previously /oved by the Commission for use at any nuclear power plant, ıe basis for a contention shall adduce material facts suffi-.ent to require reasonable minds to inquire further, and such ıcts shall be supported by affidavit or probative and reliable ›cumentary material. Affidavits shall set forth such facts as ›uld be admissible as evidence and shall show affirmatively ıat the affiant is competent to testify to the matters stated ıerein.]

. [Section 2.740(b) is amended by adding a new subparagraph ?) as follows:

?) <u>Spent Fuel Proceedings</u>. In spent fuel proceedings other ıan on the first application for a license or license amend-t to expand on-site fuel storage capacity by the use of a ıw technology not previously approved by the Commission for se at any nuclear power plant, discovery shall begin only fter the presiding officer's order ruling on the admission of ontentions and shall relate only to those matters in contro-ersy. In the presiding officer's order ruling on the admis-ion of contentions, the presiding officer shall designate a eriod, not to exceed ninety days, during which all discovery ust be completed. In addition, discovery shall be limited to wo rounds, of which the second shall relate only to responses o the first. The presiding officer shall rule on any

:overy dispute expeditiously, consistent with the above lim- ns on discovery. No specified time or period, whether scribed by rule or order, shall be extended except by con- : of all parties and upon good cause shown.

paragraph (2) of the present rule shall be renumbered as paragraph (3).]

Section 2.749a is added to read as follows:

.749a <u>Authority of presiding officer to dispose of certain issues on the pleadings</u>.

Any party may request, in writing, a decision by the pre- ing officer that all or any part of the matters [admitted tentions] involved in a spent fuel proceeding need not be rd in an adjudicatory hearing. ~~Spent-fuel-proceedings,-pur- -to-Part-50-of-this-chapter,-include-an-application-for-a ense,-or-for-an-amendment-to-an-existing-license,-filed er-January-7,-1983-and-prior-to-December-31,-2005,-to-expand -spent-nuclear-fuel-storage-capacity-at-the-site-of-a-civil- -nuclear-power-reactor,-through-the-use-of-high-density-fuel rage-racks,-fuel-rod-compaction,-the-transshipment-of-spent lear-fuel-to-another-civilian-nuclear-power-reactor-within -same-utility-system,-the-construction-of-additional-spent lear-fuel-pool-capacity-or-dry-storage-capacity,-or-by-other ans.~~

(b) A request pursuant to paragraph (a) of this section shall be deemed granted upon receipt and the presiding officer shall notify all the parties as to the date, time and location of oral argument. Such oral argument will not be scheduled until all parties have completed discovery, pursuant to 10 C.F.R. §§ 2.740-2-2.742 and 2.744, on the ~~matters-raised-in~~ [contentions affected by] the request. Fourteen (14) days prior to the date set for oral argument, each party~~,-including-the-NRC-Staff,-shall-submit,-to-the-presiding-officer,-and-shall-simulta-neously-serve-on-all-other-parties,~~ [shall file] a detailed written summary of all of the facts, data, and arguments which are known to the party at such time and upon which the party proposes to rely at the oral argument. [At the same time, each party shall file the sworn testimony or sworn written or documentary material upon which it proposes to rely.] Only facts and data in ~~the~~ [such] form ~~of-sworn-testimony-or-other-written submission~~ may be relied upon by the parties during oral argument[,] and [of the material submitted by the parties] the presiding officer shall only consider those facts and data submitted in such form.

(c) After due consideration of the oral presentation and the written facts and data presented at the oral argument, the presiding officer shall promptly by written order:

.) decide [or dismiss] all issues of law or fact not gnated for resolution in an adjudicatory hearing, setting h ~~fully-the-presiding-officer's-findings-and-conclusions,~~ a the reasons ~~or-bases~~ for them [such action]; and

(2) designate ~~any-remaining-questions-of-fact-or-law-for solution-in-an-adjudicatory-hearing~~ [in writing the specific cts that are in genuine and substantial dispute, the reasons ıy the decision of the Commission is likely to depend or the esolution of such facts, and the reason why an adjudicatory ıearing is likely to resolve the dispute.]

(d) No question of law or fact shall be designated for resolution in an adjudicatory hearing unless the presiding officer termines that:

(1) there is a genuine and substantial dispute of fact which can only be resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing; and

(2) the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute.

(e) In making a determination under paragraph (d) of this section, the presiding officer ~~shall-designate-in-writing-the-specific-facts-that-are-in-genuine-and-substantial-dispute,-the reasons-why-the-decision-of-the-Commission-is-likely-to-depend on-the-resolution-of-such-facts,-and-the-reasons-why-an-adjudicatory-hearing-is-likely-to-resolve-the-dispute.--The-presiding~~

~~officer~~ shall not ~~consider~~ [designate for adjudicatory resolution:

(1) any contention involving the same factual issue previously decided in another spent fuel proceeding, absent the existence of significant new or differing information that subtantially affects the previous decision or other good cause;

(2) any contention not supported by sworn testimony or exhibits sponsored by a qualified expert; or

(3) any contention involving no more than a differing technical judgment applied to an undisputed set of facts, where a clear consensus of the scientific community -- as established in NRC Regulatory Guides or other credible published technical reports -- controverts the technical judgment advanced in such contention.

(1[4]) any ~~issue~~ [contention] relating to the design, construction, or operation of any civilian nuclear power reactor already licensed to operate at such site, or any civilian nuclear power reactor for which a construction permit has been granted at such site, unless the presiding officer determines that any such ~~issue~~ [contention] substantially affects the design, construction, or operation of the facility or activity for which such license application, authorization, or amendment is being considered; or

(2[5]))] any siting or design ~~issue~~ [contention] fully considered and decided by the Commission in connection with the

ınce of a construction permit or operating license for a
ıan nuclear power reactor at such site, unless (i) such
: [contention] results from any revision of siting or de-
criteria by the Commission following such decision; and
the presiding officer determines that such ~~issue~~ [conten-
] substantially affects the design, construction, or op-
ːon of the facility or activity for which such license ap-
ıtion, authorization, or amendment is being considered.

~~----If-the-presiding-officer-determines-that-no-issue-is-to-be~~
~~gnated-for-an-adjudicatory-hearing,-the-order-required-by~~
~~graph-(c)-of-this-section-shall-be-in-the-form-of,-and~~
~~l-constitute-the-initial-decision-of-the-presiding-officer~~
~~ccordance-with-the-provisions-of-Section-2.760-of-these~~

]) This section shall not apply to a proceeding on the
t application for a license or license amendment to expand
ite fuel storage capacity by the use of a new technology
previously approved [by the Commission] for use at any nu-
r power plant ~~by-the-Commission.~~

OHN J. KEARNEY, Senior Vice President

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February 17, 1984

(10)

Secretary of the Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

DOCKET NUMBER PROPOSED RULE PR-2,72 (48FR 54499)

Att'n: Docketing and Service Branch

Re: 10 C.F.R. Parts 2 and 72: Proposed Rule on Hybrid Hearing Procedures for Expansions of Onsite Spent Fuel Storage Capacity at Civilian Nuclear Power Reactors

Gentlemen:

The following comments are submitted on behalf of the Edison Electric Institute (EEI) and the Utility Nuclear Waste Management Group (UNWMG). EEI is an association of the nation's investor-owned utilities; its members generate about seventy-eight percent of the nation's electricity and serve over sixty-seven million customers. UNWMG is comprised of forty-two utilities with specific interests relating to nuclear spent fuel storage. Its members are listed in Attachment A hereto. A significant number of the member utilities of EEI and UNWMG are likely to require expansion of onsite spent fuel storage prior to 1998 when the Department of Energy is committed to begin removal of spent fuel from the site of commercial nuclear power plants.

On December 5, 1983, the Commission published in the Federal Register a proposed rule that would amend its regulations at 10 C.F.R. Parts 2 and 72 to implement Section 134 of the Nuclear Waste Policy Act of 1982 (NWPA), which prescribed expedited licensing procedures for certain spent fuel storage technologies. 48 Fed. Reg. 54499 (1983). Consistent with the NWPA, the changes to existing Commission procedures would apply only to applications for a license or license amendment to expand onsite spent fuel storage capacity at commercial nuclear power reactors through the use of high-density fuel storage racks, fuel rod compaction, the transshipment of spent nuclear

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fuel to another civilian nuclear power reactor within the same utility system, the construction of additional spent nuclear fuel pool capacity or dry storage capacity, or by other means. The proposed new procedures would not apply to the first application for a license amendment to expand onsite fuel storage capacity by the use of a new technology not previously approved by the Commission for use at any nuclear power plant. Two options are identified in the proposed rule.

Option 1 substantially departs from the existing practice and would require the use of a "hybrid" hearing procedure in all proceedings to which Section 134 applies. For example, Option 1 removes the requirement that to be admitted to a licensing proceeding a petitioner must specify at least one valid "contention." Somewhat broader discovery would be allowed on any "issue" raised by intervenors and found to be within the scope of the proceeding. An "oral argument" procedure would be established as a means of determining those issues which should be adjudicated. Option 2 is a less drastic departure from existing rules and would provide a new summary disposition procedure utilizing oral argument, to be employed at the request of any party to the proceeding. As a result of the procedure in both proposed Option 1 and Option 2, the presiding officer would designate an issue for adjudication if there is a genuine and substantial dispute of fact which can be resolved with sufficient accuracy only by the introduction of evidence in an adjudicatory hearing and if the decision of the Commission is likely to depend in whole or in part on the resolution of such a dispute.

EEI/UNWMG finds that Option 1 is inconsistent with both the language of Section 134 of the NWPA and the legislative history and intent of Congress in enacting Section 134. While Option 2 is technically consistent with the wording of Section 134, it does not go as far as Congress intended in establishing meaningful procedural reform to provide an expedited proceeding for the expansion of spent fuel storage capacity at existing civilian nuclear power reactors. In this letter we propose modifications to Option 2 and additional procedures that are consistent with the Congressional mandate of Section 134.

The Legislative Purpose of Section 134 of the NWPA

Nowhere in the "Supplementary Information" published with the proposed rule nor inherent within the proposed changes to existing Commission procedural requirements does the Commission come to grips with the Congressional intent in enacting Section 134 of the NWPA. Nowhere does the Commission state its purpose in proposing changes to existing procedures other than to implement Section 134. Yet the intent of Congress in adopting Section 134 was clear, and it is just as clear that the Commission's proposal fails to accomplish what Congress intended.

The legislative history of the NWPA actually spans a period of over five years and three Congresses. During this period the utilities lobbied vigorously for a comprehensive Federal program for away-from-reactor interim storage of spent fuel. In finally passing the NWPA, the Congress did not establish the comprehensive Federal program for interim storage that the utilities had sought. Instead, Congress found that:

> [T]he persons owning and operating civilian nuclear power reactors have the primary responsibility for providing interim storage of spent nuclear fuel from such reactors, by maximizing, to the extent practical, the effective use of existing storage facilities at the site of each civilian nuclear power reactor, and by adopting new onsite storage capacity in a timely manner where practical. . . .

Section 131(a)(1) of the NWPA, 42 U.S.C. § 10151 (1983). The Congress did establish a limited Federal interim storage program to ensure that utilities did not lose full core reserve capability at the site of a nuclear reactor if diligent pursuit of onsite alternatives failed to provide in a timely manner for needed onsite storage capacity. Section 135 of the NWPA, 42 U.S.C. § 10155 (1983). But while the Congress found that utilities had the primary responsibility for spent fuel storage onsite, it also found that:

> [T]he Federal Government has the responsibility to <u>encourage and expedite</u> the effective use of existing storage facilities and the addition of needed new storage capacity at

> the site of each civilian nuclear power reactor. . . .

Section 131(a)(2) of the NWPA, supra (emphasis added). To accomplish this Congressional finding, Section 134 was adopted. Simply put, Section 134 was a trade-off; the utilities failed to convince Congress of the need for a major Federal program for interim storage of spent fuel, but Congress instead provided for expedited licensing of onsite spent fuel storage technologies.

While there is no Conference Committee report to provide a definitive legislative history of the NWPA, statements of the floor managers of the bills in the House and Senate during congressional debates and Committee Reports from the two houses leave little doubt as to the intent of the Congress in finally enacting Section 134. In the Senate, the precursor to the NWPA was S. 1662, a consensus bill drafted by members of the Senate Committee on Energy and Natural Resources and the Senate Committee on Environment and Public Works, and introduced by Senator McClure. During Senate debate of S. 1662, Senator Simpson explained to his colleagues the relationship between the proposed Federal interim storage policy in Title III of S. 1662 and the proposed changes to NRC procedures for expanding onsite spent fuel storage, as follows:

> Title III of the compromise provision establishes a firm, and I believe, appropriate national policy for the interim storage of spent fuel. Under this policy, the utility operators of nuclear powerplants bear the primary responsibility for interim storage of spent fuel at the sites of their nuclear plants. This places a significant, but appropriate, burden on the utilities to do everything possible to assure sufficient onsite storage capacity through a variety of measures specified in the legislation. These measures include reracking, transshipment of spent fuel between reactors within the same utility system, and the use of new technologies such as dry storage and the use of storage casks.

> In order to assist the utilities in carrying out this responsibility, the legislation contains measures to expedite the necessary regulatory approvals from the Nuclear Regulatory Commission for these various means of expanding onsite storage at reactor sites. These expedited licensing procedures in themselves represent an important step toward reforming this aspect of NRC's licensing process, while at the same time incorporating important safeguards to assure that the public health and safety is protected in these spent fuel storage expansion efforts.

128 Cong. Rec. S4157 (daily ed. April 28, 1982).

Senator Thurmond opposed Federal away-from-reactor spent fuel storage and offered an amendment to eliminate the provision for interim storage in S. 1662. In offering his amendment, Senator Thurmond reminded his fellow Senators that the "streamlined regulatory" process would remain:

> It should be stressed, however, that our amendment leaves intact those provisions of title III which establish a streamlined regulatory process to aid utilities in licensing additional storage space at reactor sites or in licensing transshipments of spent fuel to other sites.

128 Cong. Rec. S4274 (daily ed. April 29, 1982). Senator Simpson opposed the amendment and characterized the interim storage program in Title III as "'last resort,' emergency relief." Id. at S4281. Senator Simpson continued:

> This [national] policy [for spent fuel storage] places primary responsibility with the utilities for providing adequate spent fuel storage capacity at the reactor sites.
>
> [I]n order to carry out this element of the national policy, the bill includes new licensing procedures that are intended to expedite NRC approvals of utility requests for spent fuel storage expansion at reactor

> sites. These new procedures, which involve interim licensing authority and the use of hybrid hearing procedures, should minimize the potential for unnecessary delays in processing these utility license applications.

The Senate rejected Senator Thurmond's amendment (id. at :87) and subsequently passed S. 1662.

Section 313 of S. 1662 was similar to Section 134 of : subsequently enacted NWPA, although it did not limit issues it could be considered and did not proscribe the hybrid pro- lure in proceedings involving a new technology. During hear- js on S. 1662, Chairman Palladino testified:

> S. 1662 has a number of important provisions with which we agree. It recognizes the need for additional storage facilities for spent fuel both onsite at reactors and at separate sites away from reactors; and the need to expedite the licensing activities related to expanding spent fuel capacity onsite at a reactor.

clear Waste Disposal: Joint Hearings on S. 637 and S. 1662 fore the Senate Comm. on Energy and Natural Resources and the bcomm. on Nuclear Regulation of the Senate Comm. on Environ- nt and Public Works, 97th Cong., 1st Sess 236 (1981) (state- nt of Nunzio J. Palladino, Chairman, NRC).

The efforts by the House of Representatives to pass a nprehensive nuclear waste bill during the 97th Congress were re complicated. Three major committees -- Interior and Insu- r Affairs, Science and Technology, and Energy and Commerce -- d jurisdiction, and each reported and approved separate lls: H.R. 3809, reported in H.R. Rep. No. 491, Part 1, 97th ng., 2d Sess. (1982); H.R. 5016, reported in H.R. Rep. No. .1, Part 1, 97th Cong., 1st Sess. (1981); and H.R. 6598, ported in H.R. Rep. No. 785, Part I, 97th Cong., 2d Sess. .982). Subsequently, the three committees entered into nego- .ations to reconcile H.R. 3809, H.R. 5016, and H.R. 6598. The :sult of the negotiations was H.R. 7187. This bill was pres- ited to the House on September 30, 1982 as a substitute amend- :nt to H.R. 3809. 128 Cong. Rec. H8162 (daily ed. September

30, 1982). It was passed by the House on December 2, 1982. 128 Cong. Rec. H8800 (daily ed. December 2, 1982).

Many of the provisions of H.R. 7187, including Section 134, were drawn from H.R. 6598 as amended by the Committee on Energy and Commerce.1/ Section 134 had been added to H.R. 6598 by the Committee on Energy and Commerce; it provided for hybrid hearings in license or license amendment proceedings to expand onsite spent nuclear fuel storage capacity, restricted the issues that could be litigated in such a proceeding, and authorized interim licensing. The Committee explained:

> Procedural changes are made to the NRC licensing process to encourage utilities to expand storage capacity at reactor sites. Except for the use of a technology which has been adopted on a generic basis, each of the methods for expanding storage capacity requires a license or an amendment to the existing operating license. The bill provides for expediting the consideration of such application by "scoping" issues in an informal oral argument preceded by discovery and requiring at the conclusion of such informal oral argument that the Commission designate disputed questions of fact and law for formal adjudication only if it determines there is a genuine disputed issue of fact and the Commission's decision is likely to depend in whole or in part on the resolution of the issue(s) they seek to raise in order to be granted an adjudicatory hearing. In any Commission proceeding to expand spent fuel storage capacity, six categories of issues, such as need for power generated by the reactor involved, would be excluded from consideration. In addition the Commission is authorized to grant an interim license or interim amendment to an existing licence for expansion of onsite storage or transshipment prior

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1/ 128 Cong. Rec. H8168 (daily ed. September 30, 1982) (statement of Rep. Dingell).

> [to] the conduct or completion to any hearing required by law, provided that in all other respects the requirements of the law are met and there is assurance that public health and safety will be protected and refusal to grant an interim license would prevent a petitioner from providing adequate onsite storage capacity.

.. Rep. No. 785, Part I, 97th Cong., 2d Sess. 39 (1982).2/

H.R. 6598 as amended also authorized a limited amount Federal interim spent fuel storage; but the Committee made clear that onsite capacity was the primary means of interim rage. Similar to section 301 of S.1662 (the bill which the ate had passed), Section 131 of H.R. 6598 established the icy of "maximizing, to the extent practical, the effective of existing storage facilities at the site of each civilian lear power plant." The Committee added,

> The Federal Government is charged with the responsibility to provide limited "last resort" interim storage capacity for civilian nuclear power reactors determined by the Nuclear Regulatory Commission to be needed to assure the continued orderly operation of the reactor, through the maintenance of full core reserve storage capability.

. Rep. 785, Part 1, 97th Cong., 2d Sess. 39 (1982).

H.R. 7187 eliminated the interim licensing authority at had been included in Section 134 of H.R. 6598. H.R. 7187 so reformulated the issues that were excluded from the scope hybrid proceedings. Except for the absence of subsection)(4) on the licensing of new technology, which was added by bsequent amendment, Section 134 of H.R. 7187 was identical to subsequently enacted provision.

Representative Dingell suggested that H.R. Rep. No. 785 be nsidered part of the legislative history of H.R. 7187. 128 ng. Rec. H8168 (daily ed. September 30, 1982).

During the floor debates, the hybrid provisions were mentioned only incidentally, during discussions on an amendment to eliminate the provisions for federal interim storage; but these comments again emphasized the intent of Congress to expedite expansion of onsite storage. Rep. Lundine, the proponent of the amendment, stated

> My amendment would also preserve provisions in the bill for expedited NRC licensing provisions for at reactor interim storage. These streamlined procedures at the NRC will insure timely action on licensing issues.

128 Cong. Rec. H8581 (daily ed. November 30, 1982). Similarly, Rep. Broyhill, in opposing Rep. Lundine's amendment, stated

> Well, another purpose of the bill is this: Section 131 . . . continuing through sections 132, 133, and 134, provides for expedited consideration of applications for expansion of onsite storage of these spent fuels, and certainly there is a crying need for these expedited procedures. Generally speaking, I would say there is agreement that these expedited procedures for the licensing of these onsite facilities are needed, and if there is no final resting place by 1998, obviously there is going to have to be some consideration for the expansion of onsite storage.

Id. at H8584. Rep. Lundine's amendment was subsequently rejected. Id. at H8590.

The Senate and House bills that had been passed -- S. 1662 and the text of H.R. 7187 as H.R. 3809 -- were not referred to a House-Senate conference; instead, in order to expedite the legislation, the Senate Committee on Energy and Natural Resources introduced an amended version of the House-passed bill. 128 Cong. Rec. S15621, S15639-42, S15669 (daily ed. December 20, 1982). On December 20, 1982, the Senate passed this bill, and the House then agreed to the Senate amendments. Id. at S15670, H10525.

The Senate bill amended Section 134, but only to add a restriction on the use of the hybrid procedure; the hybrid procedure was prohibited in proceedings to expand onsite fuel storage by the use of new technology. Id. at S15643-44. However, debates on this amendment once again stressed that the purpose of Section 134 was to expedite:

> Section 134 of the McClure substitute amendment to H.R. 3809 provides for an abbreviated, legislative-type hearing to procede the normal full adjudicatory hearing. The purpose of the abbreviated hearing is to speed up the licensing of onsite storage expansion. A full hearing would only be necessary if it were determined that a "genuine and substantial dispute of fact" exists; that such dispute could be resolved in a full adjudicatory hearing; and that the decision of the Commission is likely to depend in whole or in part on the resolution of the dispute. The criteria by which the Commission may decide that a full adjudicatory hearing is necessary is extremely narrow.

Id. at S15644 (statement of Sen. Mitchell).

Thus it is indisputable that the intent of Congress in enacting Section 134 of the NWPA was to expedite the licensing of expanded onsite storage. Congress perceived an expedited licensing procedure as essential if powerplant operators were to bear the burden of supplying sufficient interim storage capacity. As we discuss below, neither Option 1 nor Option 2, as proposed by the Commission addresses this clear legislative intent.

OPTION I

EEI/UNWMG strongly opposes Option 1. The procedure set forth in Option 1 is inconsistent with the NWPA. For example, Option 1 is not optional and therefore does not comply with the statutory mandate that an opportunity for oral argument be provided "at the request of any party." 42 U.S.C. § 10154(a) (1983). Moreover, not only does the procedure proposed in Option 1 fail to satisfy the clear legislative intent

--to expedite the licensing process and hence the expansion of onsite spent fuel storage capacity -- but in our view it is likely to lengthen the process.

We provide a section-by-section analysis of Option 1 in Attachment B hereto. Briefly, we have the following specific concerns with respect to Option 1. It eliminates contention pleading requirements. Thus, discovery is wide-open to any "issue" which an intervenor may wish to raise. Indeed, the Commission admits "discovery will be somewhat broader than under existing practice." 48 Fed. Reg. at 54501. Also, Option 1 eliminates the traditional "one good contention" rule for party status. The implied corollary is that an intervenor remains a party to a proceeding even if all its allegations are summarily resolved. Finally, Option 1 permits cross-examination during the oral argument and calls for formal findings and conclusions. This procedure far exceeds "oral argument" and borders on formal adjudication; and it implies that all issues which are not designated for adjudication must be decided by the Licensing Board, whereas dismissal might be appropriate.

Presumably the benefit the Commission believes would result from the hybrid procedure proposed in Option 1 is the disposition of most if not all issues after oral argument, thus avoiding or narrowing the scope of a hearing. Yet, as noted by the Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals on the Proposed Nuclear Licensing Reform Act of 1983:

> [I]n general, the major delays associated with public hearings are attributable to the time devoted to getting to the public hearing and to the time required to obtain decisions following the public hearing. With rare exceptions, the public hearings themselves -- even with protracted cross examination -- have not been a material schedule factor in the overall public hearing process.

Report of the Ad Hoc Committee for Review of Nuclear Reactor Licensing Reform Proposals on the Proposed Nuclear Licensing Reform Act of 1983 (December 15, 1982), at 14. We submit that Option 1 would allow an intervenor intent on delay ample opportunity to bog down the process with unbridled discovery, an untried and unduly formal oral argument procedure, and the possibility of a hearing in any event on some issues after this process. Indeed, often licensees, when the schedule for obtaining a license amendment is crucial, avoid filing motions for summary disposition of issues in a proceeding -- even where the result almost surely would be favorable -- in order to avoid the delay in getting to hearing and to a decision. The unanimous judgment of attorneys representing utility members of EEI/UNWMG is that Option 1 would lengthen the process, and would thus utterly fail in achieving Congressional purpose.

OPTION 2

Option 2 is much preferable to Option 1. The normal rules for the pleading and admissibility of contentions apply, and this procedure ensures that only specific, controverted matters are referred to the hybrid procedure.3/ In addition, the one good contention rule remains in effect and allows the dismissal of intervenors who fail to advance litigable issues. Also, Option 2 makes the hybrid procedure optional, consistent with the Act, and conforms more closely to the procedures prescribed by the Act. In particular, Option 2 does not authorize cross-examination during oral argument.

On the other hand, not all of the problems in Option 1 are eliminated in Option 2. The Commission ignored the invitation from Congress to fashion expeditious rules of discovery particularly applicable to this type of proceeding. Option 2 places no time limits on discovery, and specific limits are absolutely essential if the hybrid procedure is to expedite the licensing process. *See* § 2.749a(b). Also, like Option 1, Option 2 does not provide for prefiled sworn testimony and written submissions, and the prefiling of this material would permit the parties to better prepare for oral argument. *Id.* Similarly, Option 2 does not make it clear that the "written submissions" which may be relied upon refer to sworn written or documentary material admissible as evidence. *Id.* In addition, like Option 1, Option 2 provides that the presiding officer

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3/ To the extent there may be some ambiguity in the wording of Option 2, we have proposed clarifying language which makes it clear that the oral argument procedure is available only for disposition of contentions previously admitted.

shall only consider those facts and data submitted in the form of sworn testimony or written submission. Id. This provision may inappropriately preclude official notice and is inconsistent with the Act. See discussion of Option 1, § 2.1105(b) at Attachment B. Finally, Option 2 calls for a decision supported by formal findings and conclusions on issues not designated for adjudication.

Although EEI/UNWMG finds Option 2 preferable to Option 1, it cannot support Option 2 in its present form. Option 2 merely replaces one summary disposition procedure with another (albeit with an improved standard) and therefore does relatively little to expedite the licensing process. For that reason, it ignores the clear intent of the NWPA and squanders the opportunity to develop an innovative and efficient hearing process for licensing spent fuel storage technologies. Accordingly, EEI/UNWMG strongly recommends adoption of the additional procedures discussed below. Attachment C hereto sets forth the actual text of our proposed modifications and additional provisions to be incorporated with Option 2.4/

Additional Changes to the Commission's Rules to Implement Congressional Intent in Enacting Section 134 of the NWPA

To remedy the problems discussed supra with respect to Option 2, EEI/UNWMG proposes certain changes to the proposed Section 2.749a. These changes include: 1) amending subsection (b) to clarify the evidentiary nature of "written submissions;" 2) amending subsection (b) to require the prefiling of sworn testimony and written submissions; 3) amending subsection (b)

4/ While not part of the proposed rule, EEI/UNWMG strongly endorses the Commission's strict interpretation of the "Sholly Amendment" as it applies to applications for expansions of onsite spent fuel storage technologies. See 48 Fed. Reg. at 54500, note 1. We anticipate that license amendments to permit spent fuel storage expansions generally will not involve a "significant hazard" consideration and such license amendments can be issued immediately. Thus, the expedited procedures that we proposes here are in all parties' interests, particularly where the hearing process is available only subsequent to the issuance of the license amendment.

so that official notice is not inadvertently precluded; 4) amending subsection (c) so that the presiding officer is authorized to dismiss issues; and 5) amending subsection (c) and deleting subsection (f) to eliminate formal findings. See Attachment C at 3-5.5/

EEI/UNWMG also believes that the Commission must address the question of procedural reform, consistent with Section 134, that will meaningfully expedite the licensing process.6/ In this regard, in addition to the modifications to Option 2 proposed above, we propose that Commission

(1) adopt a threshold *prima facie* test for admission of a contention;

(2) limit discovery to the scope of admitted contentions and no more than two rounds, to be completed during a period established by the presiding officer not to exceed ninety days; and

(3) establish by rule criteria to be considered by a Board, after hearing oral argument pursuant to the Option 2 procedures, in determining whether a contention should be litigated in an adjudicatory proceeding.

A. Contentions

The present rules governing admissibility of contentions, which require the party offering a contention simply to state the basis for the contention with reasonable specificity, is inappropriate for proceedings involving expansions of onsite

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5/ A number of clarifying changes have also been proposed to Section 2.749a. For example, EEI/UNWMG proposes changing the words "matters" and "issues" to "contentions," since Section 2.749a applies to admitted contentions.

6/ Such procedural reforms as we propose here may or may not be appropriate for consideration in the broader context of overall licensing reforms. That issue is not addressed here.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION II
SAM NUNN ATLANTA FEDERAL CENTER
61 FORSYTH STREET, SW. SUITE 23T85
ATLANTA, GEORGIA 30303-8931

December 28, 1999

Carolina Power & Light Company
ATTN: Mr. James Scarola
Vice President - Harris Plant
Shearon Harris Nuclear Power Plant
P. O. Box 165, Mail Code: Zone 1
New Hill, NC 27562-0165

SUBJECT: NRC INSPECTION REPORT NO. 50-400/99-12

Dear Mr. Scarola:

This refers to the inspection conducted on November 15 - 19, 1999, at your Harris facility. This was a special team inspection covering activities related to the planned expansion of the Shearon Harris spent fuel pool. The objectives of this inspection were to assess the implementation of the construction quality assurance program in construction of the C and D spent fuel pools, evaluate the alternate weld inspection program, and evaluate the plans for commissioning of the equipment for the C and D spent fuel pools (SFP).

The inspection found that CP&L had a comprehensive program to control, inspect, and document welding at the time of original plant construction in accordance with Section III of the ASME Boiler and Pressure Vessel Code, and NRC requirements. The inspection also found that the alternate weld inspection program was adequate to provide assurance that the welds for which documentation was missing, met design requirements. The program for commissioning of the C and D SFP equipment will be examined in an inspection tentatively planned for January 24 - 28, 2000. No violations of NRC requirements were identified during the inspection.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Sincerely,

Kerry D. Landis, Chief
Engineering Branch
Division of Reactor Safety

Docket No. 50-400
License No. NPF-63

Enclosure: NRC Inspection Report

cc w/encl: (See page 2)

cc w/encl:

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U. S. NUCLEAR REGULATORY COMMISSION

REGION II

Docket Nos.: 50-400

License Nos.: NPF-63

Report Nos.: 50-400/99-12

Licensee: Carolina Power & Light Company (CP&L)

Facility: Shearon Harris Nuclear Power Plant, Unit 1

Location: 5413 Shearon Harris Road
New Hill, NC 27562

Dates: November 15 - 19, 1999

Team Leader: J. Lenahan, Senior Reactor Inspector
Engineering Branch
Division of Reactor Safety

Inspectors: B. Crowley, Senior Reactor Inspector
K. Heck, Quality Assurance Engineer, NRR
D. Naujock, Materials Engineer, NRR

Approved By: Kerry D. Landis, Chief
Engineering Branch
Division of Reactor Safety

SUMMARY OF FINDINGS

Shearon Harris Nuclear Power Plant
NRC Inspection Report 50-400/99-12

The fuel pool cooling systems are described in Section 9.1.3 of the licensee's Updated Final Safety Analysis Report (UFSAR). The design basis for pools A and B, which support the operation of Unit 1, is identical to that for pools C and D. Because these pools are located in a single building and major system components needed to be installed during the early phase of construction, procurement and installation of the major system components for all four spent fuel pools was performed concurrently, in the late 1970s and early 1980s. In a letter dated December 23, 1998, the licensee requested an amendment to the Shearon Harris facility operating licensee to place spent fuel pools (SFP) C and D in service to increase the onsite spent fuel storage capacity. The licensee is currently operating and storing fuel in the A and B SFP. The majority of the C and D SFP were completed prior to 1982 during plant construction.

During preparation of the plans for completion of the C and D SPF, the licensee discovered that documentation for 52 welds on ASME Class III piping had been inadvertently destroyed. The 52 welds were 40 piping welds and 12 welded attachments for pipe hangers (lugs). The 40 piping welds included 15 spent fuel system welds which are embedded in concrete, 22 accessible spent fuel system welds, and 3 accessible component cooling system welds. Three of the accessible spent fuel system welds were subsequently removed and replaced with new welds, resulting in 37 piping welds with missing records. The most significant missing documents were the weld data reports (WDRs) for each of the welds. In order to demonstrate the weld quality for the welds with missing documentation, the licensee developed and implemented an alternative inspection program.

This special inspection included a review of the construction quality assurance (QA) and quality control (QC) program; the original construction QA/QC records; the licensee's alternative inspection program for welds with missing QA/QC records; the engineering service requests prepared to complete the C and D SFP; a walkdown inspection of the accessible C and D SPF components; and the licensee's program for commissioning of the C and D SFP. The inspectors used Temporary Instruction (TI) 2515/143 for guidance during this inspection.

The inspection found that the licensee had a comprehensive program to control, inspect, and document welding at the time of original construction in accordance with Section III of the ASME Boiler and Pressure Vessel Code, and NRC requirements. The inspection also found that the licensee's alternative weld inspection program was adequate to provide assurance that the welds for which documentation was missing, met design requirements. The licensee's program for commissioning of the C and D SFP equipment should ensure that existing equipment meets design requirements and will perform its design function. An Inspector Followup Item (IFI) was opened to inspect implementation of the equipment commissioning process. No violations were identified.

REPORT DETAILS

1. REVIEW OF THE LICENSEE'S CONSTRUCTION QUALITY ASSURANCE PROGRAM

1.1 Review of Quality Assurance and Quality Control Procedures

Inspection Scope

The inspectors reviewed Quality Assurance (QA) and Quality Control (QC) procedures that implemented the QA program requirements during construction.

Observations and Findings

The inspectors reviewed the licensee's ASME Quality Assurance Manual for the Construction of the Shearon Harris Nuclear Power Plant transmitted to NRC by letter dated dated April 30, 1999. This Manual described the quality assurance program that implemented the quality assurance requirements of ASME Boiler and Pressure Vessel Code, Section III, Division 1, Nuclear Power Plant Components, and applicable Federal, State and local regulations and codes. The Manual was applicable to fabrication and construction of ASME components which include the A, B, C and D spent fuel pools.

The inspectors reviewed the implementing QA and QC procedures listed below which controlled activities relating to weld quality. The procedures revisions were applicable to the time during 1979-1981 when the major weld activity for construction of the spent fuel pools occurred. Procedures reviewed were as follows:

Number, Revision	Title
CQA-1, Rev. 5	Personnel Training and Qualification
CQA-2, Rev. 0	QA Document Control
CQA-4, Rev. 5	QA Records
CQA-8, Rev. 3	Material Issue Surveillance
CQA-12, Rev. 0	Mechanical Equipment Installation Monitoring
CQA-14, Rev. 0	Application and Control of "N" Type Symbol Stamps
CQA-15, Rev. 0	Assignment and Control of National Board Serial Numbers
CQA-16, Rev. 0	Preparation and Submittal of ASME Code Data Reports
CQA-18, Rev. 0	Control of Site Fabrication/Modification of Piping Subassemblies
CQA-20, Rev. 0	Surveillance of Contractor Welding and Related Activities
CQA-22, Rev. 0	Welding Activity Monitoring
CQA-24, Rev. 0	Procurement Control
CQA-28, Rev. 0	QA Surveillance
CQA Appendix A	Quality Assurance Forms
CQC-2, Rev. 3	Nonconformance Control
CQC-4, Rev. 3	Procurement Control

CQC-6, Rev. 0Receiving Inspection
CQC-8, Rev. 3Storage Control

CQC-10, Rev. 0	Cleanness Control
CQC-12, Rev. 0	Mechanical Equipment Installation Control
CQC-13, Rev. 0	Concrete Control
CQC-19, Rev. 0	Weld Control
CQC-20, Rev. 0	Post-Weld Heat Treatment Control
CQC-22, Rev. 3	Hydrostatic Test Inspection
CQC-23, Rev. 0	Systems Turnover

The procedures were consistent with the CP&L QA program, established by the ASME QA Manual and NRC requirements, and defined specific process requirements in sufficient detail to provide for QA/QC control of welding activities.

A detailed review was performed for procedures CQC-19, Weld Control; CQC-22, Hydrostatic Test Requirements; and CQC-13, Concrete Control. This review was directed toward determining an alternate method to ascertain the quality of the field welds for which certain records were missing. These procedures are described below.

Weld Control

CQC-19 assigned the Welding QA/QC Specialist the responsibility for: review and verification of data and designated hold points in the Weld Data Reports (WDRs); ensuring completed WDRs for code welds were forwarded to the Authorized Nuclear Inspector (ANI) for review; supervising the QC Inspectors in the performance of weld inspections; and monitoring activities related to welding. QC inspection personnel were trained and qualified in accordance with CQA-1. The SFP field welds, which were ASME Code Class 3 welds, were documented on a WDR , reviewed and approved by the Welding QA/QC Specialist, and reviewed for acceptance by the ANI. The ANI performed an independent third party review. The responsibilities of the Welding QA/QC Specialist and QA inspection personnel were sufficiently defined to provide reasonable assurance that the quality of the completed field welds were in compliance with applicable ASME Code requirements. After the documentation of a field weld was determined to be acceptable, pertinent documents were assembled and the package was transmitted to QA Records in accordance with CQA-4.

Hydrostatic Test Inspection

CQC-22 established the requirements for performing hydrostatic test inspections to ensure that hydrostatic tests were performed in accordance with approved procedures and specifications. The Mechanical QA Specialist was responsible for verifying that the documentation for the piping was completed prior to performance of the hydrostatic test. This included verification that field welds within the scope of a hydrostatic test had been satisfactorily completed, inspected, and accepted. The Mechanical QA Specialist was also responsible for performance of the leak inspection during hydrostatic testing. QC inspection personnel also witnessed the test. The responsibilities of the Mechanical QA Specialist and QC inspection personnel were sufficiently defined to provide assurance that the quality of hydrostatic testing was in compliance with applicable procedures and specifications. After the documentation for a hydrostatic test had been accepted by the ANI, the pertinent documents were assembled and reviewed by the Mechanical QA Specialist, who verified that manufacturing/fabrication records for components within the boundaries of the test had been received and accepted and that there were no open nonconformances on any of the components.

Concrete Placement

CQC-13 and Construction Procedure WP-05, Concrete Placement, established the requirements for assuring all work activities in the area affected by a concrete pour were completed prior to placement of concrete. A prerequisite to placement of concrete was the completion of a Concrete Placement Report, which signified that all activities in the affected area had been satisfactorily completed such that access to the area to be covered with concrete was no longer required. When specific crafts completed their work, the appropriate Craft Superintendent signed off the Concrete Placement Report, signifying that a particular activity, such as mechanical, electrical, cadwelds, nondestructive examination, or cleanup, was complete and ready for the concrete pour. This sign-off was required by all Craft Superintendents, whether or not they had work in the particular placement, as a safeguard against omissions. After sign-off by the Craft Superintendents, Field Engineering signed the Concrete Placement Report, verifying that required design attributes, such as the correct location and anchoring of embedded conduit, grounding, inserts, sleeves, piping, and plumbing, were complete and correct. When all the crafts had completed their work, the Construction Inspector signed the report, signifying that all work had been inspected and approved. Subsequently, Quality Control and Quality Assurance signed the report signifying that all of their oversight activities were completed and that the items to be embedded in the concrete were in compliance with applicable requirements. Finally, after all required disciplines, QA, Construction Inspector and design approval sign-offs were completed, the Area Superintendent authorized concrete placement activities to proceed. The completed Concrete Placement Report was transmitted to QA Records in accordance with CQA-4.

Conclusions

The QA/QC procedures in effect at the time of construction of the SFP provided comprehensive control of welding and other construction activities. The procedures provided holdpoints to assure welding was completed in accordance with ASME and NRC requirements prior to proceeding beyond a point wherein any nonconformances could be resolved. These included a detailed review of weld documentation to assure the welds were completed in accordance with technical requirements, and that the welds were inspected and tested prior to being subjected to a hydostatic pressure test. For welds which were to be embedded in concrete, completion of the Concrete Placement Report provided an additional holdpoint to assure the welds were satisfactory prior to placement of concrete. The ANI provided an independent third party review of the ASME welding program.

1.2 Review of Welding Process Control Procedures

Inspection Scope

The inspectors reviewed original construction welding process control procedures, which were in effect at the time the existing Fuel Pools "C" and "D" equipment and piping were installed, as detailed below.

Observations and Findings

The welding control procedures listed below were reviewed to verify that a quality assurance program was in place at the time of installation of Fuel Pools "C" and "D" piping to ensure that pipe welding was accomplished in accordance with applicable Code requirements. The procedure revisions were those applicable when the welding activities for the fuel pools were in progress. Procedures reviewed were as follows:

MP-01, Revisions 3, 5, 6, and 7, Qualifying of Welding Procedures

MP-02, Revision 4, Procedure for Qualifying Welders and Welding Operators

MP-03, Revisions 1, 3, and 4, Welding Material Control

MP-06, Revisions 3, 4, and 5, General Welding Procedure for Carbon Steel Weldments

MP-07, Revisions 3 and 4, General Welding Procedure for Stainless Steel Nickel Base and Nonferrous Weldments

MP-09, Revisions 1, 9, and 10, Welding Equipment Control

MP-10, Revisions 2 and 3, Repair of Base Materials and Weldments

MP-11, Revisions 3, 4, and 5, Training and Qualification of Metallurgical/Welding Engineering and Support Personnel

MP-12, Revisions 1, 2, and 3, Control of Special Welding Materials for BOP and Welding Material for Non-Permanent Plant

MP-13, Revisions 1 and 2, Welder Qualification for Areas of Limited Accessibility

The procedures provided detailed control for all aspects of the welding process, including qualification of procedures and welders, control of welding materials, control of welding variables, and quality documentation for each weld.

Conclusions

At the time of original construction of the existing fuel pool cooling system piping, a comprehensive welding program was in place to control and document pipe welding in accordance with Section III of the ASME Boiler and Pressure Vessel Code.

2. REVIEW OF CONSTRUCTION QA/QC RECORDS

2.1 Review of Hydrostatic Test Reports

Inspection Scope

The inspectors reviewed the records documenting the results of hydrostatic testing performed on the piping welds embedded in the C and D fuel pool concrete.

Observations and Findings

The inspectors reviewed the records which documented completion of hydrostatic testing in accordance with WP-115 and the licensee's quality assurance program. Records examined were for the following C and D fuel pool embedded piping welds numbers : 2-SF-1-FW-1, -2, -4, & -5; 2-SF-149-408; 2-SF-143-512, 513, & -514; 2-SF-144-FW-515, -516, & -517; and 2-SF-159-FW-518 & -519. These records were documented on CP&L form QA-26, pages one and two of two, Hydrostatic Test Records. Information on the data sheets included the hydrostatic test boundaries (welds tested), the piping design pressure, test pressure, the test medium and test temperature, test data, and the test results. The test prerequisites required that the mechanical QA specialist verify that all required piping documentation was completed, and that all required weld documentation was completed. The inspectors verified that the hydrostatic test records specified that all weld records were completed, and that the welds were accepted by the quality assurance group prior to start of the hydrostatic test. The inspectors also verified that the records had been signed by the ANI. The hydrostatic test records for the above welds showed that all welds were tested to a minimum of 25 percent above design pressure and that all welds met the test acceptance criteria. The licensee did not retain copies of the form QA-26 for embedded weld numbers 2-SF-8-FW-65 & -66. However, in response to questions during construction regarding hydrostatic testing of the welds attaching the liner plate to the piping spool pieces, the licensee initiated Deficiency and Disposition Report (DDR) 794. Resolution of this DDR included documentation of the dates various welds were hydrostatically tested. The dates the welds for piping spool pieces were hydrostatically tested (July 19, 1979 and July 24, 1979) were listed in the DDR response. These included weld numbers 2-SF-8-FW-65 & 66. The inspectors concluded that the documentation for DDR-794 provided evidence that weld numbers 2-SF-8-FW-65 & 66 were subjected to hydrostatic testing in accordance with WP-115 and the licensee's quality assurance program.

Conclusions

The hydrostatic test records documented that the embedded welds were subjected to hydrostatic testing, and met the test acceptance criteria. The records also provided evidence that the welds were completed, inspected and documented in accordance with the licensee's quality assurance program. The hydrostatic test records provide evidence that the WDRs were reviewed prior to performance of the hydrostatic tests.

2.2 Review of Concrete Placement Reports

Inspection Scope

The Inspectors reviewed the concrete placement records for spent fuel pools C and D which documented that all work and preparations for the concrete placements were completed and that all required inspections had been completed prior to placement of concrete.

Observation and Findings

Prior to placement of concrete, a concrete placement report was completed to document that all work activities have been completed in a particular area (slab, column, wall, etc) and that the concrete placement could proceed. The inspectors reviewed drawing numbers SK A-G-0126, South Fuel Pool Area of FHB Isometric, and SK A-G-0125, FHB Isometric North Fuel Pool Units 2 & 3, to determine the concrete placement numbers which contained the embedded piping for the C and D fuel pool cooling system. This review showed that the piping had been installed in the following C & D fuel pool placement numbers: wall placements W-255-7, W-261-7, -7A, -9, -10, and -11, W-281-10, -16, -17, and -18, and slab placements SL-246-3 and SL-246-4. The inspectors reviewed the placement report for the above listed placement numbers and verified that the placement reports had been properly completed and signed prior to placement of concrete. The inspectors verified that the mechanical embed/piping had been signed in accordance with CP&L procedure WP-05. The acceptance criteria noted on the placement reports for mechanical embed/piping was CP&L procedure WP-102, Installation of Piping. Procedure WP-102 required that a verification be performed to assure that all piping was installed as per the design drawings. Additional requirements referenced by procedure WP-102 were that hydrostatic testing of piping to be embedded in concrete was to be completed in accordance with CP&L procedure WP-115, Hydrostatic Testing of Buried or Embedded Piping.

Conclusions

The concrete placement reports provide evidence that the piping embedded in the concrete was inspected and tested in accordance with the requirements of the licensee's construction quality assurance program prior to concrete placement. These requirements included verification that the welding was completed in accordance with applicable procedures, and that documentation such as WDRs were completed and reviewed prior to the concrete placement.

2.3 Review of ASME Documentation

Inspection Scope

The Inspectors reviewed completed documentation required by the ASME Boiler and Pressure Vessel Code for the fuel pool cooling systems.

Observation and Findings

10 CFR 50.55, "Codes and standards," requires that systems and components of pressurized water-cooled nuclear reactors meet certain requirements of the ASME Boiler and Pressure Vessel Code. The fuel pool cooling systems for for SFP A, B, C, and D are classified as ASME Code Section III, Division 1, Class 3 systems. The applicable edition of the ASME code is Section III, 1974, Winter 1976 Addenda.

Subsection NA of Section III addresses "General Requirements"; Subsection ND addresses requirements for "Class 3 Components". Subsection NA-8420, "Report Form for Field Installation," required that installation welds be verified on Data Form N-5, which includes attestation of the quality of the weld process and specification data for the weld filler material. The weld process was witnessed at several specified check points by a Quality Assurance inspector; the Authorized Nuclear Inspector had the option to witness any check point and verified the completed weld data report prior to closure.

The licensee's amendment request, submitted by letter dated December 23, 1998, states that certain records, notably piping isometric packages for field installation of the completion portion of SFP C and D, were inadvertently discarded. Subsection NA-8416, "Piping Systems" of the Code requires completion of N-5 forms for each piping system, which includes weld data records attesting to the quality of the weld process and weld material certification. Because these records have been lost, the SPF C and D cannot be certified as an N-stamp system.

Since piping welds for SFP A and B were completed during the same time frame as those for SFP C and D, and by the same group of welders, it is reasonable to expect similar quality of the N-5 data packages for both units. Therefore, the N-5 package for Pools A and B were examined. The N-5 forms were included as part of the N-3 package, which was submitted upon completion of Unit 1 to the ASME National Board, the enforcement authority having jurisdiction. The N-3 form listed the components including interconnecting welds and the data reports for a facility. The summary N-3 package for Unit 1 was examined by the inspectors.

Subsection NA-8400 identifies the reporting requirements for various components, including valves and pumps, parts and appurtenances, pipe subassemblies, and piping systems. Only the reporting requirements for 49 field welds cannot be met. The inspectors randomly selected data packages for two C and D SFP components: a pump (2B-SB) and a strainer (3-SF-53-5A-2). The data package for the pump included a Certificate of Compliance, a Manufacturer's Data Report (NPV-1), material certification, hydrostatic test reports, performance test reports, welding ticket records, dimensional inspection records, a cross-sectional drawing, and an as-built drawing. The data package for the strainer included an ASME Code data report, a Certificate of

Conformance, liquid penetrate reports, a product quality control check list, material test reports, an inspection and test report, dimensional inspection records, and sequence traveler.

Conclusions

The ASME N-3 and N-5 data packages for Unit 1 and the ASME data packages for two SPF C and D components reviewed by the inspectors were determined to be complete and satisfactory and provided an indication that the licensee documented construction of the SFP in accordance with ASME requirements.

2.4 Review of Audits of ASME QA Program Implementation

Inspection Scope

The inspectors randomly selected an audit of ASME QA program implementation for review.

Observations and Findings

CP&L corporate audits were conducted of the ASME QA Program Implemented at Shearon Harris. The inspectors retrieved a listing of these audits from the licensee's data base and noted that eight such audits had been conducted during the period from March 19, 1979 through February 19, 1982. From these audits, the inspectors randomly selected audit QAA/170-6 for review. QAA/170-6 was conducted at the Shearon Harris site on September 21-29, 1981. The inspectors reviewed the audit checklist, the audit report containing the findings and concerns, the memoranda describing the corrective actions for each identified deficiency, and the QA closure documentation. The audit report concluded that the Shearon Harris Construction, Nuclear Plant Engineering, and QA Program adequately met ASME code requirements except for eleven findings and sixteen concerns. The identified deficiencies were typically associated with procedural and training requirements and indicative of careful review by the auditors. The inspectors reviewed the corrective actions and found them reasonable and appropriate. All corrective actions were implemented and determined to be satisfactory by the licensee'sQuality Assurance organization within four months following the audit.

Conclusions

The audit report showed that the licensee's QA program implemented the ASME program and NRC requirements during construction.

2.5 Review of Vendor ASME QA Program Implementation

Inspection Scope

The inspectors reviewed an audit of a vendor supplying Code equipment for compliance with ASME requirements.

Observations and Findings

The inspectors reviewed CP&L corporate audit QAA/702-1, conducted at the fabrication facility of Southwest Fabricating & Welding Company, Inc., a supplier of piping spool pieces for the four spent fuel pools at Shearon Harris. The audit was conducted on May 22-23, 1974, in order to appraise the the manufacturing facility and quality assurance program to adherence to purchase order requirements, including applicable Articles of Section III of the ASME Boiler and Pressure Vessel Code and the requirements of 10 CFR 50, Appendix B, "Quality Assurance for Nuclear Power Plants." The audit report concluded that the vendor's quality system, as defined in its QA Manual was adequate to meet the intent of the requirements imposed by the purchase order. The audit report identified six findings requiring corrective action. The inspectors reviewed the audit checklist and the audit report containing the findings. The inspector also reviewed the corrective actions taken by the vendor and the QA closure documentation. Based on this review, the inspectors determined that the deficiencies were relatively minor and administrative in nature and that the corrective actions were appropriate. All actions were determined to be satisfactory by the CP&L Quality Assurance organization within three months of the audit with exception of an issue related to training and qualification of audit personnel. This issue was held open pending resolution of a related draft ANSI standard and closed satisfactorily in December, 1974.

Conclusions

The vendor audit report showed that the licensee's QA program implemented the ASME program and NRC requirements for performance of vendors during construction.

2.6 Review of QA/QC Related Reports

Inspection Scope

The inspectors reviewed a random sample of QA/QC related reports to assess the effectiveness of the site QA/QC program in identifying and resolving problems associated with SFP welding activities.

Observations and Findings

Reports documenting results of QA/QC activities were reviewed by the inspectors to assess the effectiveness of the QA/QC program. The reports selected for review covered the period when welding activities were in progress on the piping from 1979 to 1982. The records reviewed include Deficiency and Disposition Reports (DDRs), Nonconformance Reports (NCRs), and QA/QC monitoring and surveillance reports. DDRs for ASME Code components required the ANI to review, approve and sign the final disposition as acceptable. The following DDRs, which are listed in general categories assigned by the inspectors, were reviewed:

Category	DDR

Arc Strike	869, 877, 895, 945
Stamping	888, 889, 914, 945
Holdpoint	829, 1009
Hydrostatic Test	783, 794

The identified deficiencies were clearly identified on the DDR and disposition of the deficiencies were appropriate. Concurrence with the disposition by the ANI and report closure by Quality Assurance was completed for all DDRs reviewed.

Nonconformances (NCRs) were less significant infractions of the QA program requirements (i.e., were less serious than DDRs). The following NCRs were reviewed and listed in general categories assigned by the inspectors.

Category	NCR
Arc Strike	WP-206
Stamping	W-027, W-096, W-103
Holdpoint	W-207
Welder Requirement	WP-111, W-028
Weld Status Report	WP-278

Documentation of the nonconforming condition was clear and corrective actions were appropriate. The final disposition for each NCR was verified by the responsible QA Specialist.

For completeness of review, the inspectors arbitrarily selected a sample of QA/QC reports which documented monitoring and surveillance of weld activities. These covered areas which included material control, welding equipment, welder training and qualification, review of WDRs for accuracy and completeness, and compliance with weld procedures. The following QA/QC activity reports were reviewed and determined to be typical and expected for oversight of welding activities.

WP62, WS79, WP56, W29, W86, W116, W124, W143, W199, W200, W285, W297, W322, W361, W365, W402, W429, W434, W456, W461, W462, W469, W475, QA8, QA81, WS80, QA146, QA150, QA189, QA215, QA294, QA359, QA424, QA368, QA376, QA509, QA548, QASRC83116, QA550, QA551, QA586, QA587, QA588, QA703, QA777, W509, W507, W506, W503, W767, W756, W750, QA16, QA254, QASRC187, QASRC822660, QA199, W630, W560, W554, W544, W519, W518, QA385, W8257, W225.

Conclusions

Based on review of the above DDRs, NCRs, and reports documenting QC/QA activities, the inspectors concluded that inspection personnel actively monitored welding activities and processes for compliance with ASME Code and QA Program requirements. Deficiencies were accurately reported, corrective actions promptly taken, and appropriately resolved. All corrective action documents reviewed were in compliance with the licensee's QA program and NRC requirements.

3. SFP C AND D DESIGN CHANGES

Inspection Scope

The inspectors reviewed the design changes prepared by licensee engineers to complete the C and D spent fuel pools.

Observations and Findings

The licensee implements design changes in accordance with CP&L procedure EGR-NGGC-0005, Engineering Service Requests (ESR). This procedure implements the design control program required by 10 CFR 50, Appendix B. The licensee prepared the following ESRs to complete the C and D spent fuel pools:

- ESR 95-00425, Study Effort to Support Fuel Pool in Service Date.

- ESR 99-00218, CCW Tie In to Heat Exchangers for North Pools

The inspectors reviewed the ESRs. ESR 99-00218 was prepared for connecting the C and D spent fuel pool heat exchangers to the Unit 1 component cooling water system. During the inspection, the licensee was in the process of installing piping and pipe supports required for the tie-in of the CCW system to the SFP C and D heat exchangers. The final tie in will not be completed unless NRC approval is received for the fuel pool expansion. ESR 95-00425 was prepared to complete the C and D SFP piping, complete installation of equipment (pump motors, strainers, etc.), perform system pre-operational and startup testing, and revise existing plant procedures to incorporate the C and D SFP into the Unit 1 operating plant.

The inspectors reviewed the 10 CFR 50.59 safety evaluation, design inputs, design evaluations, assumptions, and references, design verification documentation, and installation drawings and instructions. The inspectors noted that the details for commissioning of the existing equipment were incomplete. The licensee initiated ESR 99-00416 to control the commissioning process. This is discussed in the Section below. The requirements and procedures for preoperational and startup testing were also incomplete. Discussions with licensee engineers disclosed that these procedures will be developed following those used for startup of Unit 1 (SFP A and B). The 10 CFR 50.59 evaluation concluded that this project involved an unreviewed safety question which required NRC approval prior to completion and startup.

Conclusions

The ESRs were technically adequate and generally met regulatory requirements.

4. EQUIPMENT COMMISSIONING

Inspection Scope

The inspectors examined the licensee's maintenance and lay-up actions for the installed Fuel Pool "C" and "D" piping and equipment. In addition, plans for additional activities to ensure that equipment will meet all applicable requirements and be capable of performing its intended function were reviewed.

Observations and Findings

A significant portion of the Fuel Pool Cooling System and Component Cooling Water System piping and components for Fuel Pools "C" and "D" were installed during original construction in the late 1970s and early 1980s. As documented in section 26.5.0 of Engineering Service Request (ESR) Design Specification 95-00425, Revision 0, the equipment was never incorporated into the operating unit and has not been formally maintained under controlled storage since that time. The equipment was procured and installed to applicable quality assurance requirements. However, since the installed equipment has been stored in-place without a formal storage and lay-up program, the licensee plans to implement an equipment commissioning or dedication process to ensure that the equipment will meet the applicable requirements and is capable of performing its intended function in the completed design. In accordance with ESR 95-00425, which had not been approved and issued at the time of the inspection, a Matrix of Commissioning Requirements is to be developed, which will define the requirements, including any additional inspections and testing, for each component. At the time of the inspection, a preliminary matrix had been developed as part of ESR 95-00425 and ESR 99-00416 had been initiated to further detail and manage the commissioning process. Although plans and some of the details for the process were included in ESR 95-00425, most of the details for each individual component were still being developed to be included in ESR 99-00416. Based on discussions with responsible licensee personnel and review of ESR 95-00425, the commissioning process will consist of the following activities:

Scope Development

To develop the scope for the commissioning process, a field walkdown of the installed equipment (mechanical, civil, instrumentation and control, and electrical) will be performed to compare the installed equipment with the completed modification design and each item in scope will be identified and individually dispositioned as part of ESR 99-00416.

Document Review

Quality documentation will be retrieved and reviewed to ensure that required quality assurance information is available, complete and acceptable. The verified records will include original procurement and field installation records. The equipment installation records will be compared with field conditions to ensure that the installation as accepted has not been altered. If records are missing or deficient, an assessment will be performed to determine what can be accepted by virtue of retest or re-inspection, or by use of alternate methods of verification.

Test and Acceptance Criteria

The Equipment Commissioning Matrix will specify additional activities needed to ensure the required level of quality assurance because of the lack of formal storage and lay-up program since original equipment installation. These activities will include:

Field verification of equipment identification against procurement documentation with establishment of traceability to Code Data Reports for code related equipment.

Physical inspections and testing as required to verify that lack of controlled storage conditions and regular maintenance has not caused any condition (corrosion, aging, etc.) adverse to quality.

Physical inspections and considerations necessary to ensure that plant activities since construction have not resulted in any conditions adverse to quality (scavenging of parts, introduction of foreign material, damage from personnel and equipment traffic, etc.).

Although the equipment commissioning details for individual equipment had not been finalized, some work had already been accomplished. The inspectors reviewed the following work requests (WRs) that had been issued:

WR 98-AGAR1 - Disassemble and Inspect Valve 1CC-512
WR 98-AFJA1 - Inspect Train A Spent Fuel Cooling Heat Exchanger
WR 98-AFJE1 - Inspect Train B Spent Fuel Cooling Heat Exchanger
WR 98-AFJF1- Disassemble and Inspect Train A Spent Fuel Cooling System Strainer
WR 98-AFJH1- Disassemble and Inspect Train B Spent Fuel Cooling System Strainer
WR 98-AFIY1- Disassemble and Inspect Spent Fuel Pool Cooling Pump 2A
WR 98-AFIZ1- Disassemble and Inspect Spent Fuel Pool Cooling Pump 2B

Disassembly and inspection had been completed for WRs 98-AGAR1, 98-AFJA1, 98-AFJE1, 98-AFJH1. The other 3 WRs had not yet been worked. For inspection of the Heat Exchangers, the WRs only covered removing the end covers and inspecting the tube side of the Heat Exchangers. The WRs indicated that a nitrogen purge had been maintained on the shell side of the heat exchangers. However, further investigation revealed that the use of the nitrogen purge had not been implemented until late 1991. In May of 1988, WRs 88-AMYH1 (Train A) and 88-AMYI1 (Train B) were issued to provide a nitrogen purge on the shell side of the Heat Exchangers. The WRs documented that the shell side of the Heat Exchangers had been open to the Fuel Building atmosphere. There was no indication how long the heat exchangers had been open. The 1988 WRs installing the purge were not worked until December 1991. Also, additional WRs documented a number of problems with low nitrogen purge on Train B Heat Exchanger in 1993. Based on the documented history of lack of control of the atmosphere on the shell side of the Heat Exchangers, the Inspectors questioned whether additional evaluations of the Heat Exchangers were needed. In response, the licensee indicated that further evaluations of the shell side of the Heat Exchangers will be performed as part of the commissioning process under ESR 99-00416.

The inspectors walked down and observed the general condition of the installed piping and equipment. Even though the equipment had not been maintained under a formal program, the equipment and piping appeared to be well preserved. The inspectors also examined spent fuel pool cooling pump motors "A" and "B", which have been stored and maintained in the warehouse since procurement at the time of construction. These were found to be in good condition with the motor space heaters energized. Evidence of control of storage of the pumps, including records of periodic pump shaft rotation, maintenance of heat on motors, and megger testing, were reviewed. Preventative maintenance of these parameters had been maintained in accordance with licensee Material Evaluation Procedure ME 000261.03.

The inspectors inspected three welds, weld numbers 2-CC-3-FW-207, 2-CC-3-FW-208, and 2-CC-3-FW-209 for misalignment and concluded that there was no noticeable misalignment.

The inspectors reviewed the re-inspection records for installed welds and piping as discussed below.

Based on the above reviews, the inspectors concluded that the planned equipment commissioning process should ensure that existing equipment will meet requirements and will perform its design function. However, since the details of tests and inspections to be performed for individual equipment items had not been completed, Inspector Followup Item (IFI) 50-400/99-12-01, Review of Final Equipment Commissioning Details, was opened to track further inspection after more details are available.

Conclusions

Although details of the commissioning inspections had not been finalized for each individual piece of equipment, a detailed plan had been drafted and if properly implemented should ensure that existing equipment meets requirements and will perform its intended function. An IFI was opened to track further inspection of the equipment commissioning process after more details of the tests and inspections to be performed for individual equipment items are available. The equipment commissioning WRs reviewed were considered appropriate to ensure that equipment is acceptable to place in service. Based on the documented history of lack of control of the atmosphere on the shell side of the Spent Fuel Pool Cooling Heat Exchangers, the inspectors concluded that additional evaluations of the heat exchangers were needed.

5. ALTERNATE INSPECTION PROGRAM

5.1 Review of Weld Records

Inspection Scope

The inspectors reviewed the Spent Fuel Cooling System and Component Cooling System weld and weld inspection records as detailed below.

Observations and Conclusions

The licensee re-inspected all existing accessible Fuel Pool "C" and "D" Spent Fuel Pool Cooling System (SFPCS) and supporting Component Cooling Water System (CCWS) pipe and pipe attachment field welds. The welds were visually (VT) and liquid penetrant (PT) inspected. In addition, vibro-tooled welder symbol identifications were taken from each weld surface and welder qualification verified by review of records. The re-inspections and the welder symbols were documented on new Weld Data Reports (WDRs). The inspectors reviewed the new WDRs, the NDE qualification records for the current re-inspections and the original construction welder qualification records for these welds. All records were retrievable and found to be in order.

In addition to review of the re-inspection records for the accessible welds, records consisting of WDRs, welder qualification records, weld QC inspector records, NDE examiner qualification records, welding procedure specifications (WPSs), and procedure qualification records (PQRs) were reviewed for the below listed Unit 1 SFPCS piping welds. These Unit 1 (SFP A and B) welds were constructed using the same welding QC program at approximately the some time period as that used for the cooling system piping welds for Fuel Pools "C" and "D".

F1-236-1-SF-10-FW-60
F1-236-1-SF-2-FW-9
F1-236-1-SF-10-FW-58
F1-236-1-SF-2-FW-8
F1-236-1-SF-10-FW-59
F1-236-1-SF-2-FW-6
F1-236-1-SF-2-FW-7

These original Unit 1 (SFP A and B) construction records were retrievable, legible, and complete. The records provided objective evidence that a detailed welding quality control program was in place and followed during original construction.

Conclusions

All records reviewed were retrievable and in order. The original Unit 1 construction records provided good assurance that the SFP C and D welding was accomplished and documented in accordance with the approved welding quality assurance program in effect at that time.

5.2 Welding Material

Inspection Scope

The inspectors reviewed the welding procedure specifications and the records for the filler metal (materials) used for welding the SFPCS and CCWS piping.

Observations and Findings

SFP A & B Filler Metal

The inspectors randomly selected embedded SFPCS welds from isometrics drawings, 1-SF-2 and 1-SF-10 from SFP A and B for review. The WDRs for these welds were reviewed by the inspectors. From the WDRs, the inspectors randomly selected the certified material test reports (CMTRs) for filler and insert metals and reviewed the chemical test records. Based on the records reviewed, the inspectors concluded that the materials used for the embedded welds were type 308 filler metal, type 308 consumable inserts, and type 304 base material (piping materials).

The inspectors reviewed Weld Procedure Specification (WPS)1BA3 for the material used for welding the pipes in the component cooling water system. The WPS listed the pipe material as P-1, Grade 1 (Appendix D to Section XI of the ASME Code) and weld filler metals as E70S-6 and E7018. For procedure qualification, WPS 1BA3 referenced Procedure Qualification Report (PQR) 15. The inspectors reviewed PQR 15 and CMTRs of the material used for the qualifications.

Product Check Chemistries

The inspectors compared the chemistries from CMTRs with the stainless steel product check chemistries submitted to NRC in a letter dated April 30, 1999, Subject: Response to NRC Request for Additional Information Regarding The Alternative Plan for SFPCS Piping, and the chemical analyses from PQR 15 that were used for qualifying the carbon steel weld procedure specification 1BA3 with product check chemistries submitted to NRC in a letter dated June 14, 1999. The comparisons showed carbon analyses for the product checked consistently above the filler metal values for SFP A & B and values recorded in the PQR. The inspectors questioned the licensee regarding possible carbon contamination with the product check chemistries.

In search of the contamination, the inspectors examined the sampled surface on weld 2-CC-3-209. The sample had been removed from the center of the weld crown. The weld and surrounding pipe were clean and free of foreign matter. Next, the inspectors reviewed the technique used for sampling. The sampling technique is in Appendix A to Procedure NW-16, Revision 1, "Identification of Base Metals for Welding Applications," dated January 6, 1998. The sampling technique uses a rotary carbide deburring tool which removes material with a grinding action. Licensee engineers suspected that the deburring tool was a possible source of the carbon contamination. The licensee made test samples by taking known material and seeding it with metal flakes broken from the teeth of the deburring tool. The tests showed that for samples seeded with 5 and 10 weight percent from the deburring tool, the carbon analyses increased by .03 and .08 weigh percent, respectively. The tests showed that the carbide deburring tool was a possible source of carbon contamination.

Alloy Comparator

During the inspection, the inspectors witnessed a demonstration of the test method used to develop the acceptance criteria for the test data submitted to NRC in the April 30, 1999 letter. For the testing, the licensee utilized the Metorex X-Met 880 electronic unit, CP&L Control No. MLCE-132 which was operated by CP&L's plant metallurgist. The inspectors reviewed the following: Operating Instruction Manual 3881 432-4VE; and operating procedure: MCP-NGGC-0101, Revision 1, Test Method 4, dated March 26, 1999. For developing an acceptance criteria, the metallurgist setup the X-Met using the same calibration and reference standards that were used for the previous testing. For calibration, pure standards for Fe, Cr, Ni, Cu, Mo, and a backscatter sample were run and stored in the X-Met. For reference alloys, stainless steel standards for type 304, 309, 310, 316, and NIST C1154a were run and stored in the X-Met reference library.

For the development of the acceptance criteria, 12 different standards were used. Each standard was run 10 times producing an average set of chemical values. In the comparison mode, the X-Met compared each test against the standards stored in the reference library. If the test matched or was close to a match with a reference standard, the X-Met displayed the reference standard followed by the term: good, possible, or good/possible. If a test did not come close to any reference standard, the X-Met displayed "no good match." The reference standards, test standards, type of match displayed for that standard, and the Cr, Ni, Mo, Mn, and Cu from the certified analysis reports for the standards are shown in Table 1 in the Appendix. The data showed that the X-Met comparison mode can discriminate stainless steel types and chemical extremes within a stainless steel type. Based on the testing performed on the accessible field welds and Table 1, the licensee's metallurgist tentatively established the acceptance criteria for field welds as two test displays showing a good or possible match and no test displays showing no good match.

Conclusions

The SFPCS piping and CCW piping was welded using the correct materials. The X-Met and chemical analysis provided identification of stainless steel and carbon steel materials.

5.3 Water Quality

Inspection scope

The inspectors reviewed the C & D SFP pipe welds exposed internally to hydrostatic pressure test water and/or the spent fuel pool water.

Observations and Findings

The inspectors reviewed drawings and hydrostatic test records to identify the C & D SFP welds that were exposed internally to hydrostatic pressure test water or spent fuel pool water, to determine the length of time that these welds were exposed to that water. Of the 52 welds identified in CP&L's letter dated April 30, 1999, pipe welds 2-SF-1-FW-3, 2-SF-1-FW-6, and 2-SF-36-FW-448 were replaced by new welds, and 12 are hanger-to-pipe welds. Of the remaining 37 pipe welds with missing documentation, the inspectors identified 15 welds exposed to hydrostatic test water, 22 welds exposed to the fuel pool liner leak test water, and the same 22 welds exposed to the current fuel pool water conditions.

Hydrostatic test water quality was specified in CP&L Procedure WP-115, Revision 0, "Hydrostatic Testing of Buried or Embedded Pressure Piping," dated September 19, 1979. WP-115 specified that potable or lake water was to be used for hydrostatic testing. After testing, the procedure required that the pipes must be drained. However, the procedure did not specify a time limit for draining of the piping/system. The inspectors were unable to determine from documentation when the piping was drained. However, logic dictates that the pipes were drained before the licensee performed the fuel pool liner leak testing (hydrostatic test).

Hydrostatic test water quality for fuel pool liners was identified in CP&L Procedure TP-57, "Hydrostatic Test of Fuel Pool Liners," dated May 17, 1983. TP-57 required that that the fuel pool be leak tested for a 24 hour period using unchlorinated site water. The procedure defined unchlorinated water as site water with a chloride content not exceeding 100 parts per million (ppm). After the test, the procedure required that the test water was pumped out of the SFP and that the pool was rinsed with demineralized or distilled water. Attachment A to TP-57 for SFP D showed that the pool was filled June 11, 1985 with water containing less than 1 ppm chlorides and that the rinse was completed on November 1, 1985. For SFP C, the records showed that the pool was filled May 7, 1985 with water containing less than 1.5 ppm chlorides and that the rinse was completed on November 4, 1985.

Discussions with licensee engineers disclosed that SFPs C & D were filled with SFP quality water around 1989 and have been full ever since. The gates between SPF A and B and C and D were opened at various times which resulted in the water mixing between the pools. During April 1999, the licensee obtained water samples from the low points in seven of eight pipe lines connected to SFP C & D. These samples were analyzed for impurities. The results are tabulated in Table 2 in the Appendix. The inspectors compared the sample results to the administrative limits for A & B SFP and data for a primary system cold shut down that is published in NUREG CR-5116, Survey of PWR Water Chemistry, February 1989. Based on the data reviewed, the water quality in SFP C & D was similar to the water quality in SFP A and B.

The pipe welds exposed to the potentially poorest water quality were the embedded welds. If corrosion or fouling were to occur, they would occur in the embedded welds first. The presence of corrosion or fouling would be visible from the interior of the piping. The visual inspection of the embedded welds performed by the licensee to examine the interior of the embedded piping is discussed below.

Conclusions

The pipe welds exposed to the potentially poorest water quality were the 15 embedded welds. The pipe welds remaining were exposed to treated water with very low impurities and similar to the water quality in SFP A and B. If corrosion or fouling were present in the SFP C and D piping, they would occur in the embedded welds first because of the type of water the embedded piping was exposed to.

5.4 Review of the Procedure for Remote Visual Inspection of Welds and Piping

Inspection Scope

The procedure used for remote visual inspection of embedded welds was examined for compliance with the CP&L Quality Assurance Program and NRC requirements.

Observations and Findings

The inspectors reviewed Temporary Procedure SPP-0312T, Temporary Procedure For Remote Visual Examination of Interior Welds and Surfaces of Embedded Unit 2 Spent Fuel Pool Cooling Piping for C and D Pools. The procedure provided instructions for performing remote visual examinations of interior welds and surfaces of embedded piping for the SFP C and D piping. The results of these examinations were used to determine whether the weld quality and interior surface conditions meet the acceptance criteria established in Paragraph 6.0 of the procedure. The acceptance criteria specified that welds were to be free of the following defects: cracks, lack of fusion, lack of penetration, oxidation ("sugaring"), undercut greater than 1/32 inch, reinforcement ("push through") exceeding 1/16 inch, concavity ("suck back") exceeding 1/32 inch, porosity greater than 1/16 inch, or inclusions. Any recordable indications of these defects were recorded on Attachment 1 of the procedure. Other indications such as arc strikes, foreign material, mishandling, pipe mismatch, pitting and microbiologically induced corrosion were also recorded on the attachment and were required to be evaluated by licensee engineers.

In addition to reviewing SPP-0312T, the following referenced documents were examined by the inspectors with respect to applicable requirements: (1) ASME Section III, 1974, Subsection ND-4424, Surfaces of Welds; NDEP-0606, Rev. 4, Remote Visual Examination; NDEP-601,Rev. 13, VT Visual Examination of Piping System and Component Welds at Nuclear Power Plants; and NDEP-A, Rev. 13, Nuclear NDE Procedures and Personnel Processes.

Both Revision 0 (approved 5/17/99) and Revision 1 (approved 9/9/99) of procedure SPP-0312T were reviewed. Revision 1 contained no change in the technical content or scope of work, but was made to reflect a new vendor and contract number. Based on review of the procedure and applicable references, the inspectors determined that the procedure prescribed prerequisites, precautions and limitations, and detail on special tools and equipment to adequately control the scope of the visual inspection activities. Technical, process-related, and administrative references were adequate and complete. The acceptance criteria were appropriately detailed such that conclusions as to the weld quality and interior surface conditions could be made by qualified inspection personnel. The remote inspection procedure was reviewed for adequacy prior to its use by a licensee NDE Level III Inspector. The licensee's Level III NDE inspector was interviewed by the inspectors. The Level III certification records and training for this individual were also reviewed.

Conclusions

The procedure which specified the method for visual inspection of the embedded welds provided detailed instructions and acceptance criteria for inspecting and evaluating the embedded welds. The procedure complied with the licensee's QA program and NRC requirements.

5.5 Remote Visual Examination

Inspection Scope

The inspectors reviewed the videotape that recorded the remote visual examination and the analysis of the remote visual examination of embedded welds. The review included piping and other welds captured on videotape. The inspectors also reviewed the licensee's evaluations of the welds documented on Attachment 1 to SPP-0312T.

Observation and Findings

The licensee performed a remote enhanced visual examination of 15 embedded field welds from inside the stainless steel SFP C and D piping. Prior to performance of the remote video examinations of the embedded piping, three Level II NDE personnel were trained in the use of procedure SPP-0312T. These individuals demonstrated their proficiency with the use of this procedure to the ANI and the Level III NDE inspector. Attestations to the satisfactory completion of these activities were reviewed by the inspectors and determined to be satisfactory.

The visual examination was performed by sending a mobile video camera with focusing and magnifying capabilities through the piping to examine each embedded field weld. The video camera sent images of the weld to a television monitor and video recorder. The images on the monitor were viewed by the licensee's Level II qualified remote visual inspectors. The Level II's observations were documented on Attachment 1 to SPP-0312T, "Remote Visual Examination Data Sheets." Attachment 1 contained a check list for recordable condition of the weld. These recordable conditions are described in the acceptance criteria of SPP-0312T. Weld acceptability was determined by the qualified Level II visual examiner in accordance with the acceptance criteria specified in procedure SPP-0312T and approved by a qualified Level III NDE inspector and the ANI.

The inspectors reviewed eight videotapes recorded during the remote visual inspection and the completed SPP-0312T Attachment 1 for each embedded field weld. The videotapes reviewed were as follows: weld 2-SF-8-FW-65 prior to cleaning; the in-process cleaning of 2-SF-144-FW-516; and the 15 embedded field welds after cleaning. The videotapes also captured images of accessible welds 2-SF-150-412 and 2-SF-148-FW-382.

In the videotape made prior to cleaning, the inspectors observed laced material particles inside the pipes and on the field welds. These particles looked like a dusting of snow flakes. They were flat, very thin, interconnected, and conformed to the contour of the pipes, pipe seams, and field welds. The inspectors viewed the videotape showing removal of the particles from welds 2-

SF-144-FW-516. The particles were removed with a pressurized water flow directed toward the pipes, interior surfaces. When the particles were hit by the water stream, they were readily dispersed. After dispersing, the particles appeared to be suspended in the water.

Based on the videotapes of the cleaned field welds, the inspectors concurred with the observations of the licensee's NDE inspectors recorded on the Attachment 1 to SPP-0321T for each weld. The inspectors observed the images of vendor fabricated welds, pipe seam welds, and the piping itself as the video camera traveled to the different embedded field weld locations. These images showed no misalignment, unusual protrusions, blockages, or indentations in the pipe walls, pipe seams, vendor fabricated welds, and the two accessible field welds examined. In the videotapes made of the cleaned welds, the inspectors identified conditions in three welds that require further evaluations. These conditions were: (1) an insert segment with the letters 308L still visible on weld 2-SF-144-FW-516; (2) brown spots that were out of focus with the surface of the pipe on weld 2-SF-144-FW-517, and (3) heavy stains, oxides, and deposits on weld 2-SF-159-FW-519. Although not part of the weld inspection, the inspectors also observed and requested an evaluation of a condition adjacent to the longitudinal seam in the pipe just beyond weld 2-SF-144-FW-515. The condition appears to be a fine saw tooth line located parallel to the pipe seam and about half the seam thickness away. The length of the line was not determined. The licensee stated that they were evaluating these conditions which were identified on the SPP-0312T, Attachment 1.

The inspectors reviewed and found satisfactory work requests associated with preparation for remote video inspection, and the system closure following completion of the visual inspection. These were WR/JO 99-ADUN2, ADUP1, AEHH2, and AFEY1. Results of the visual examinations were recorded on a data sheet, marked as a QA Record, which was included in SSP-0312T as Attachment 1. The data sheet was reviewed by the inspectors and determined to provide adequate detail of the examination to determine whether the acceptance criteria had been met and to record any recordable conditions noted by the licensee's NDE inspector. Completed data sheets documenting examination of 15 interior welds and piping surfaces were examined and determined to contain sufficient detail as to the results of the inspection. The signature of the NDE Level II examiner on Attachment 1 was determined to be one of the three personnel who were trained and qualified in the use of this procedure.

The recordable conditions documented on the data sheet are required to be reviewed and approved by licensee engineers and subsequently be approved by an ANI. The licensee initiated ESR 99-00266 to evaluate the recordable conditions. The evaluations were being performed by an independent engineering consultant. At the time of the inspection, evaluation of the recordable conditions had not been completed.

The inspectors reviewed and discussed the videotape examination of weld 2-SF-144-FW-516 with a CP&L welding supervisor that worked as a welding engineer during the construction of the SFP. The videotape showed the section of a consumable insert in the weld with the lettering 308L still visible on the consumable insert. The welding supervisor stated that the type of consumable insert for this application is shaped like the cross section of an inverted mushroom. The stem of the insert forms the base of the joint between the pipes. The joint is hand welded using a gas shielded tungsten arc welding process. The process should consume the insert and adjacent pipe during the first weld pass. The supervisor stated that insufficient heat input may fuse the insert (mushroom) head to the weld puddle instead of melting the insert completely. After the first pass, subsequent passes were made with filler metal to form weld layers. The supervisor estimated that 5 layers of filler metal were necessary to weld 3/8-inch thick piping.

The inspectors requested that the licensee provide chemical analysis on the particulate that were dispersed during the pipe/weld cleaning process. This particulate appeared reddish brown in color, is easily disturbed, and is believed by the licensee to be the source of the pipe stain. The inspectors questioned the ANI regarding the particulate. The ANI stated that there he observed abundant amounts of reddish brown color on the video equipment, piping interior, and at the video equipment entry point during the inspection. The licensee radiologically analyzed by chemical elements the particulate in 1990 and again in 1996. They provided the analyses to the inspectors for review. The particulate is radioactive with the most abundant element by two orders of magnitude being iron, followed by one order of magnitude cobalt, and zero order of magnitude nickel.

Conclusions

The condition of the embedded welds and associated piping inside the C and D SFP piping are free of abnormal obstructions and deposits. However, the inspectors identified four conditions requiring further evaluations. The licensee is in the process of evaluating the data shown on SSP-312T, Attachment 1 that include these four conditions.

5.6 QA Programs for Special Inspections Associated with the Alternate Inspection Program

Inspection Scope

The inspectors reviewed the alternate inspection activities for compliance with quality assurance requirements.

Observations and Findings

Ongoing activities associated with the alternate inspection program for resolution of issues concerning activation of Pools "C" and "D" were reviewed. These activities include remote inspection of the inner surfaces and field welds for embedded piping, determination of water chemistry during the period of layup, and examination of weld material taken from accessible field welds.

Oversight and examination of the embedded piping was performed by qualified NDE Level II examiners, who demonstrated proficiency in the use of the procedure used for the inspection (SPP-0312T) to the satisfaction of a NDE Level III examiner. The demonstration was witnessed and an Authorized Nuclear Inspector concurred with the demonstration of this proficiency.

Water chemistry analysis was performed by the CP&L chemistry organization, in accordance with site and corporate quality assurance program requirements. Material analysis of the weld samples was performed by NSL Analytic Services, identified on the CP&L Approved Supplier List with Supplier Control No. 16; manual dated 6/30/99; reviewed by CP&L 11/4/99. The supplier was audited for compliance under the CP&L Commercial Grade Survey program on February 1-2, 1999.

Conclusions

Activities associated with special inspections related to activation of fuel pools C and D were performed in compliance with applicable quality assurance requirements.

6. AUTHORIZED NUCLEAR INSPECTOR

Inspection Scope

The inspectors interviewed the authorized nuclear inspector (ANI) to determine the involvement of the ANI with the WDR, hydrostatic tests, and remote visual examinations.

Observations and Findings

The inspectors interviewed the recently retired ANI (July 1, 1999) and current ANI. The retired ANI was involved in plant construction and reviewed WDRs during plant construction. The verification was performed in two stages. The first stage was the verification of field weld fabrication at randomly selected predetermined hold points and ASME Code required inspection points. When satisfied that ASME requirements were met, the ANI initialed the associated line entry on the WDR. The second stage was verification of the entire WDR. When satisfied that all the necessary entries for the specified field weld were complete, the ANI signed off the WDR.

When questioned by the Inspectors regarding the significance of the ANI signature on the hydrostatic test document, both ANIs stated that the signature meant that the hydrostatic test satisfied ASME Code requirements, and the signature on the hydrostatic test was independent of any ANI signatures on the WDRs.

The ANIs were questioned regarding the extent of their involvement with the remote visual examinations of the 15 embedded welds in the C & D SFPs. They stated they both observed the equipment demonstration and qualifications of the remote visual examiners. For the equipment demonstration, a video camera was mounted on a transporting device that moved through a mockup of the SFP piping. The mockup contained flaws similar to those described in the acceptance criteria of Procedure SSP-0312T. In the mockup demonstration, the video camera transmitted images to a television monitor as it was moved. By viewing the monitor, the licensee's remote visual examiner directed the equipment operator to the areas of interest. These images were analyzed by the examiner. The examiner had to determine if the images of interest were a flaw, the type of flaw, and the acceptability of the flaw. The successful detection of flaws in the mockup demonstrated the equipment and remote visual examiner's skills. Upon a successful demonstration, the remote visual examiner qualification was certified by the licensee and verified by the ANI. On June 30, 1999, both ANIs signed off on the qualifications of the three remote visual examiners.

The inspectors questioned the current ANI regarding his involvement with the reinspection of the accessible welds and remote video examination of the embedded welds. The ANI stated that he observed the reinspection of accessible welds, 2-SF-36-FW-450 and 2-SF-38-FW-451, and that he observed the remote video inspections of at least two of the embedded welds. The actual examinations of the other embedded welds were less extensively viewed. At the time of the inspection, the ANI was in the process of reviewing the videotapes and verifying the data recorded on the remote visual examination data sheets.

Conclusions

The ANIs performed an independent verification of ASME Code requirements on the WDR and hydrostatic test documentation. The verification is part of their duties that are required by the 1974 Edition (and later) of ANSI/ASME Code N626.0, "Qualifications and Duties for Authorized Nuclear Inspection," and the referenced edition and addenda of Section III of the ASME Code. The ANIs were actively involved with the demonstration of the remote visual examination equipment and the qualification of the personnel. The current ANI was actively involved with examination and videotaping of the embedded welds

7. NRC INSPECTIONS DURING THE CONSTRUCTION PHASE

The Inspectors reviewed NRC Inspection Reports which documented inspection of construction activities by NRC Region II Inspectors between 1978 and 1983. This was the period when the A, B, C, and D spent fuel pools were under construction. The inspection reports document more than 50 separate inspections for this period for items related to the welding program and/or piping installation. The majority of these inspections were performed by eight Region II Welding Specialist inspectors. Several violations dealing with the general subject of welding were identified in these reports. Most of these violations were relatively minor (Severity Level V and VI) and would not be cited under the current NRC reactor inspection program. These violations would typically be resolved through the licensee's corrective action program. The violations were typical of what one would expect for oversight of a large construction project and are not indicative of any programmatic weakness in the licensee's welding program.

MANAGEMENT MEETINGS

The Team Leader discussed the progress of the inspection with licensee representatives on a daily basis and presented the results to members of licensee management and staff at the conclusion of the Inspection on November 19, 1999. The licensee acknowledged the findings presented.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

D. Alexander, Manager, Regulatory Affairs
B. Altman, Manager, Major Projects Section
E. Black, Level III NDE Examiner

G. Brovette, ANI
B. Clark, General Manager, Harris Plant
E. Dayton, ANI (Retired)
J. Eads, Supervisor, Licensing and Regulatory Programs
S. Edwards, SFP Activation Project Manager
G. Kline, Manager, Harris Engineering Support Services
J. Scarola, Vice President, Harris Plant
K. Shaw, Licensing Engineer, Major Projects Section
M. Wallace, Senior Analyst, Licensing
Daniel W. Brinkey III, CP&L Metallurgist
Charlie Griffith, CP&L Welding Supervisor

Other licensee employees contacted included engineering, maintenance and administrative personnel.

NRC:

R. Hagar, Resident Inspector
K. Landis, Chief, Engineering Branch, Division of Reactor Safety

INSPECTION PROCEDURE USED

TI 2515/143, Shearon Harris Spent Fuel Pool ("C" and "D") Expansion

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-400/99-12-01	IFI	Review of Final Equipment Commissioning Details

Closed

None

Discussed

None

APPENDIX 1

TABLES

Table 1

X-Met 880 Alloy Analyzer Data for Developing an Acceptance Criteria

Standard	Cr	Ni	Mo	Mn	Cu	Good/Possible Match: Alloy	No Good Match	Overall Rating
Type 304	18.28	8.13	0.17	1.48	0.19	7 / 3: Type304	----	Good
Type 309	22.60	13.81	---	1.63	---	9 / 1: Type309	----	Good
Type310	24.87	19.72	0.16	1.94	0.11	5 / 5: Type310	----	Good
Type 316	16.74	10.07	2.06	1.44	0.11	Not Analyzed	----	----
NIST C1154a	19.31	13.08	0.068	1.44	0.44	10 / 0: C1154a	----	Good
Standards Used to Check the Alloy Analyzer								
NIST 1267	24.14	0.29	---	0.315	---	0 / 0	10	No Match
NBS 1219	15.64	2.16	0.164	0.42	0.162	0 / 0	10	No Match
NBS C1289	12.12	4.13	0.82	0.35	0.205	0 / 0	10	No Match
BCS 331	15.20	6.26	---	0.78	---	0 / 0	10	No Match
NIST C1151a	22.59	7.25	0.79	2.37	0.385	0 / 0	10	No Match
NIST C1153a	16.70	8.76	0.24	0.544	0.226	0 / 9: Type304	1	Possible
NIST C1152a	17.76	10.86	0.44	0.95	0.097	0 / 4: Type304	6	No Match

NIST 1155	18.45	12.18	2.38	1.63	0.169	0 / 8: Type316	2	Possible
NIST C1287	23.98	21.16	0.46	1.66	0.58	0 / 8: Type310	2	Possible
NBS 1230	14.80	24.20	1.18	0.64	0.14	0 / 0	10	No Match
NBS C1288	19.55	29.30	2.63	0.83	3.72	0 / 0	10	No Match
NBS 1246	20.10	30.80	0.36	0.91	0.49	0 / 0	10	No Match

Table 2

Current Water Assay for C & D SFP Piping Systems, Administrative limits for A & B SFP, and NUREG CR-5116 Data for Primary Water in Cold Shut Down (ppb = parts per billion)

Identification	F (ppb)	Cl (ppb)	SO_4 (ppb)	pH
2-SF-75	57	29.5	1027	6.33
2-SF-74	29.3	62.7	682	5.82
2-SF-49	166	48	632	5.60
2-SF-215	11.7	26	321	5.55
2-SF-214	14.2	31.5	430	5.40
2-SF-212	120	70.5	676	6.74
2-SF-213	13.1	28.2	424	5.33
A & B SFP Admin. Limits (1)	<150	<150	----	----
Primary Water(2) Shut Down	<150	<150	----	----

(1) HNP Plant operating manual, Volume 5, Part 3, "SHNPP Environmental and Chemistry Sampling and Analysis Program," January 20, 1999.
(2) Shut down values above those indicated should be corrected before reaching full power operations.

AEC

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

No. H-252
Tel. 973-3335 or
973-3446

FOR IMMEDIATE RELEASE
(Monday, November 22, 1965)

AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The Atomic Energy Commission is seeking comment from the nuclear industry and other interested persons on proposed general design criteria which have been developed to assist in the evaluation of applications for nuclear power plant construction permits.

The proposed criteria have been developed by the AEC regulatory staff and discussed with the Commission's Advisory Committee on Reactor Safeguards (ACRS). They represent an effort to set forth design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such, they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It is recognized that further efforts by the AEC regulatory staff and the ACRS will be necessary to fully develop these criteria. However, the criteria as now proposed are sufficiently advanced to submit for public comment. Also, they are intended to give interim guidance to applicants and reactor equipment manufacturers.

The development and publication of criteria for nuclear power plants was one of the key recommendations of the special Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures.

In the further development of these criteria, the AEC intends to hold discussions with organizations in the nuclear industry and to issue from time to time explanatory information on each criterion. Following such discussions with industry and receipt of other public comment, the AEC expects to develop and publish criteria that will serve as a basis for evaluation of applications for nuclear power plant construction permits.

(more)

It is recognized that additional criteria may also be needed, particularly for reactors other than water reactors, and that there may be instances where one or more of the presently proposed criteria may not be applicable. Application of the criteria to a specific design continues to involve a considerable amount of engineering judgment.

These proposed criteria are part of a longer-range Commission program to develop criteria, standards and codes for nuclear reactors, including identification of codes and standards that industry will be encouraged to undertake. The ultimate goal is the evolution of industry codes based on accumulated knowledge and experience, as has occurred in various fields of engineering and construction.

A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached. Comments should be sent to the Director of Regulation, U. S. Atomic Energy Commission, Washington, D. C. 20545, by February 15, 1966.

#

11/22/65

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached hereto are general design criteria used by the AEC in judging whether a proposed nuclear power facility can be built and operated without undue risk to the health and safety of the public. They represent design and performance criteria for reactor systems, components and structures which have evolved over the years in licensing of nuclear power plants by the AEC. As such they reflect the predominating experience to date with water reactors but most of them are generally applicable to other reactors as well.

It should be recognized that additional criteria will be needed for evaluation of a detailed design, particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Moreover, there may be instances in which it can be demonstrated that one or more of the criteria need not be fulfilled. It should also be recognized that the application of these criteria to a specific design involves a considerable amount of engineering judgment.

An applicant for a construction permit should present a design approach together with data and analysis sufficient to give assurance that the design can reasonably be expected to fulfill the criteria.

FACILITY

CRITERION 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

CRITERION 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

CRITERION 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

REACTOR

CRITERION 4

The reactor must be designed to accommodate, without fuel failure or primary system damage, deviations from steady state norm that might be occasioned by abnormal yet anticipated transient events such as tripping of the turbine-generator and loss of power to the reactor recirculation system pumps.

CRITERION 5

The reactor must be designed so that power or process variable oscillations or transients that could cause fuel failure or primary system damage are not possible or can be readily suppressed.

CRITERION 6

Clad fuel must be designed to accommodate throughout its design lifetime all normal and abnormal modes of anticipated reactor operation, including the design overpower condition, without experiencing significant cladding failures. Unclad or vented fuels must be designed with the similar objective of providing control over fission products. For unclad and vented solid fuels, normal and abnormal modes of anticipated reactor operation must be achieved without exceeding design release rates of fission products from the fuel over core lifetime.

CRITERION 7

The maximum reactivity worth of control rods or elements and the rates with which reactivity can be inserted must be held to values such that no single credible mechanical or electrical control system malfunction could cause a reactivity transient capable of damaging the primary system or causing significant fuel failure.

CRITERION 8

Reactivity shutdown capability must be provided to make and hold the core subcritical from any credible operating condition with any one control element at its position of highest reactivity.

CRITERION 9

Backup reactivity shutdown capability must be provided that is independent of normal reactivity control provisions. This system must have the capability to shut down the reactor from any operating condition.

CRITERION 10

Heat removal systems must be provided which are capable of accommodating core decay heat under all anticipated abnormal and credible accident conditions, such as isolation from the main condenser and complete or partial loss of primary coolant from the reactor.

CRITERION 11

Components of the primary coolant and containment systems must be designed and operated so that no substantial pressure or thermal stress will be imposed on the structural materials unless the temperatures are well above the nil-ductility temperatures. For ferritic materials of the coolant envelope and the containment, minimum temperatures are NDT + 60°F and NDT + 30°F, respectively.

CRITERION 12

Capability for control rod insertion under abnormal conditions must be provided.

CRITERION 13

The reactor facility must be provided with a control room from which all actions can be controlled or monitored as necessary to maintain safe operational status of the plant at all times. The control room must be provided with adequate protection to permit occupancy under the conditions described in Criterion 17 below, and with the means to shut down the plant and maintain it in a safe condition if such accident were to be experienced.

CRITERION 14

Means must be included in the control room to show the relative reactivity status of the reactor such as position indication of mechanical rods or concentrations of chemical poisons.

CRITERION 15

A reliable reactor protection system must be provided to automatically initiate appropriate action to prevent safety limits from being exceeded. Capability must be provided for testing functional operability of the system and for determining that no component or circuit failure has occurred. For instruments and control systems in vital areas where the potential consequences of failure require redundancy, the redundant channels must be independent and must be capable of being tested to determine that they remain independent. Sufficient redundancy must be provided that failure or removal from service of a single component or channel will not inhibit necessary safety action when required. These criteria should, where applicable, be satisfied by the instrumentation associated with containment closure and isolation systems, afterheat removal and core cooling systems, systems to prevent cold-slug accidents, and other vital systems, as well as the reactor nuclear and process safety system.

CRITERION 16

The vital instrumentation systems of Criterion 15 must be designed so that no credible combination of circumstances can intefere with the performance of a safety function when it is needed. In particular, the effect of influences common to redundant channels which are intended to be independent must not negate the operability of a safety system. The effects of gross disconnection of the system, loss of energy (electric power, instrument air), and adverse environment (heat from loss of instrument cooling, extreme cold, fire, steam, water, etc.) must cause the system to go into its safest state (fail-safe) or be demonstrably tolerable on some other basis.

ENGINEERED SAFEGUARDS

CRITERION 17

The containment structure, including access openings and penetrations, must be designed and fabricated to accommodate or dissipate without failure the pressures and temperatures associated with the largest credible energy release including the effects of credible metal-water or other chemical reactions uninhibited by active quenching systems. If part of the primary coolant system is outside the primary reactor containment, appropriate safeguards must be provided for that part if necessary, to protect the health and safety of the public, in case of an accidental rupture in that part of the system. The appropriateness of safeguards such as isolation valves, additional containment, etc., will depend on environmental and population conditions surrounding the site.

CRITERION 18

Provisions must be made for the removal of heat from within the containment structure as necessary to maintain the integrity of the structure under the conditions described in Criterion 17 above. If engineered safeguards are needed to prevent containment vessel failure due to heat released under such conditions, at least two independent systems must be provided, preferably of different principles. Backup equipment (e. g., water and power systems) to such engineered safeguards must also be redundant.

CRITERION 19

The maximum integrated leakage from the containment structure under the conditions described in Criterion 17 above must meet the site exposure criteria set forth in 10 CFR 100. The containment structure must be designed so that the containment can be leak tested at least to design pressure conditions after completion and installation of all penetrations, and the leakage rate measured over a suitable period to verify its conformance with required performance. The plant must be designed for later tests at suitable pressures.

CRITERION 20

All containment structure penetrations subject to failure such as resilient seals and expansion bellows must be designed and constructed so that leak-tightness can be demonstrated at design pressure at any time throughout operating life of the reactor.

CRITERION 21

Sufficient normal and emergency sources of electrical power must be provided to assure a capability for prompt shutdown and continued maintenance of the reactor facility in a safe condition under all credible circumstances.

CRITERION 22

Valves and their associated apparatus that are essential to the containment function must be redundant and so arranged that no credible combination of circumstances can interfere with their necessary functioning. Such redundant valves and associated apparatus must be independent of each other. Capability must be provided for testing functional operability of these valves and associated equipment to determine that no failure has occurred and that leakage is within acceptable limits. Redundant valves and auxiliaries must be independent. Containment closure valves must be actuated by instrumentation, control circuits and energy sources which satisfy Criterion 15 and 16 above.

CRITERION 23

In determining the suitability of a facility for a proposed site the acceptance of the inherent and engineered safety afforded by the systems, materials and components, and the associated engineered safeguards built into the facility, will depend on their demonstrated performance capability and reliability and the extent to which the operability of such systems, materials, components, and engineered safeguards can be tested and inspected during the life of the plant.

RADIOACTIVITY CONTROL

CRITERION 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

CRITERION 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

CRITERION 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate hold-up capacity must be provided for retention of gaseous, liquid, or solid effluents.

CRITERION 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

COMPARISON OF DRAFTS DATED OCTOBER 20, 1966, AND FEBRUARY 6, 1967

FOR

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

February 6, 1967

Chap 9
FN 12

NOTES

1. In this comparison, the draft of October 20, 1966, is the datum.
2. Deletions made on the October 20, 1966, draft are indicated by brackets with a line through the words; e.g., [~~THE CONTAINMENT SYSTEM~~].
3. Additions to the October 20, 1966, draft are indicated by underlining; e.g., THE CONTAINMENT SYSTEM.

KEY TO CRITERIA COMPARISON

New Criteria 2/6/67	Old Criteria 10/20/66
1	1.1
2	1.2
3	2.1
4	3.2
5	-
6	2.2
7	2.3
8	3.1
9	3.2
10	3.3
11	3.4
12	3.5
13	3.6
14	3.7
15	4.0
16	4.1.1
17	4.1.1
18	-
19	4.1.2
20	4.1.3
21	4.2
22	4.3
23	5.2
24	-
25	5.1
26	-
27	5.3
28	5.4
29	6.1
30	-
31	6.2
	6.2.1
	6.2.2
32	-
33	7.0
	7.1
34	9.0

New Criteria 2/6/67	Old Criteria 10/20/66
35	9.1.2
36	7.4
37	9.1.1
	9.1.4
	9.1.5
38	7.5
39	7.2
	7.3
40	9.2.2.1
41	9.2.2.2
42	9.2.2.3
43	9.2.2.4
44	8.1
45	8.2
46	8.3
47	9.1.1
48	9.1.3
49	9.2.1.1
50	9.2.1.2
51	9.2.1.3
52	9.2.1.4
53	9.2.4.1
54	9.2.4.2
55	9.2.4.3
56	9.2.4.4
57	9.2.3.1
58	9.2.3.2
59	9.2.3.3
60	9.2.3.4
61	10.1
62	10.2
63	10.3
64	10.4
65	11.0
	11.1
	11.2

I. QUALITY AND PERFORMANCE STANDARDS

CRITERION 1 - QUALITY STANDARDS (Category A)

THOSE FEATURES OF REACTOR FACILITIES WHICH ARE ESSENTIAL TO THE PREVENTION OF ACCIDENTS WHICH COULD AFFECT THE PUBLIC HEALTH AND SAFETY OR TO MITIGATION OF THEIR CONSEQUENCES SHALL BE DESIGNED, FABRICATED, AND ERECTED TO QUALITY STANDARDS THAT REFLECT THE IMPORTANCE OF THE SAFETY FUNCTION TO BE PERFORMED. A SHOWING OF SUFFICIENCY AND APPLICABILITY OF STANDARDS IS [SHALL BE] REQUIRED. WHERE GENERALLY RECOGNIZED CODES OR STANDARDS ON DESIGN, MATERIALS, FABRICATION, AND INSPECTION ARE APPLICABLE, THEY SHALL BE USED. WHERE ADHERENCE TO SUCH CODES OR STANDARDS DOES NOT SUFFICE TO ASSURE A QUALITY PRODUCT IN KEEPING WITH THE SAFETY FUNCTION, THEY SHALL BE SUPPLEMENTED OR MODIFIED AS NECESSARY.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

THOSE FEATURES OF REACTOR FACILITIES WHICH ARE ESSENTIAL TO THE PREVENTION OF ACCIDENTS WHICH COULD AFFECT THE PUBLIC HEALTH AND SAFETY OR TO MITIGATION OF THEIR CONSEQUENCES SHALL BE DESIGNED, FABRICATED, AND ERECTED TO PERFORMANCE STANDARDS THAT WILL ENABLE THE FACILITY TO WITHSTAND, WITHOUT LOSS OF THE CAPABILITY TO PROTECT THE PUBLIC, THE ADDITIONAL FORCES THAT MIGHT BE IMPOSED BY NATURAL PHENOMENA SUCH AS EARTHQUAKES, TORNADOES, FLOODING CONDITIONS, WINDS, ICE, AND OTHER LOCAL SITE EFFECTS. THE DESIGN BASES SO ESTABLISHED SHALL REFLECT: (a) APPROPRIATE CONSIDERATION OF THE MOST SEVERE OF THESE NATURAL PHENOMENA THAT HAVE BEEN RECORDED FOR THE SITE AND THE SURROUNDING AREA AND (b) AN APPROPRIATE MARGIN FOR WITHSTANDING FORCES GREATER THAN THOSE RECORDED TO REFLECT UNCERTAINTIES ABOUT THE HISTORICAL DATA AND THEIR SUITABILITY AS A [illegible] BASIS FOR DESIGN

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 3 - REACTOR CORE DESIGN (Category B)

[THE-FACILITY-SHALL-BE-DESIGNED-WITH-MULTIPLE-BARRIERS-TO-THE-ACCIDENTAL RELEASE TO-THE-ENVIRONS-OF-FISSION-PRODUCTS.--THE-DESIGN-SHALL-INCLUDE-THE-FOLLOWING AS-BASIC-OBJECTIVES] THE REACTOR CORE SHALL BE DESIGNED TO FUNCTION [A-CORE CAPABLE-OF-FUNCTIONING] THROUGHOUT ITS DESIGN LIFETIME, WITHOUT EXPERIENCING DAMAGE THAT WOULD RESULT IN SIGNIFICANT RELEASE OF FISSION PRODUCTS FROM THE FUEL [IN-QUANTITIES-WHICH-WOULD-PRECLUDE-CONTINUED-OPERATION-BY-CONSIDERATIONS OF-10-CFR-20]. THE CORE DESIGN, TOGETHER WITH RELIABLE PROCESS AND DECAY HEAT REMOVAL SYSTEMS, SHALL PROVIDE FOR THIS CAPABILITY UNDER ALL EXPECTED CONDITIONS OF NORMAL OPERATION WITH APPROPRIATE MARGINS FOR UNCERTAINTIES AND FOR TRANSIENT SITUATIONS WHICH CAN BE ANTICIPATED; SUCH AS, THE LOSS OF POWER TO RECIRCULATION PUMPS, TRIPPING OUT OF A TURBINE GENERATOR SET, ISOLATION OF THE REACTOR FROM ITS PRIMARY HEAT SINK, AND LOSS OF ALL OFFSITE POWER.

CRITERION 4 - SUPPRESSION OF POWER AND PROCESS OSCILLATIONS (Category B)

THE CORE DESIGN, TOGETHER WITH RELIABLE PROCESS CONTROLS, SHALL ENSURE THAT POWER OR PROCESS OSCILLATIONS ARE NOT POSSIBLE OR CAN BE READILY SUPPRESSED.

CRITERION 5 - OVERALL POWER COEFFICIENT (Category B)

THE CORE, TOGETHER WITH ITS COOLING AND MODERATING SYSTEMS, SHALL BE DESIGNED SO THAT THE OVERALL POWER COEFFICIENT IN THE POWER OPERATING RANGE WILL NOT BE POSITIVE.

CRITERION 6 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

THE REACTOR COOLANT PRESSURE BOUNDARY SHALL BE DESIGNED TO HAVE THE CAPABILITY OF FUNCTIONING THROUGHOUT DESIGN LIFETIME WITHOUT FAILURES LEADING TO SIGNIFICANT LEAKAGE. [A-HIGH-INTEGRITY-COOLANT-BOUNDARY-WITH-A-CAPABILITY-TO FUNCTION-THROUGHOUT-DESIGN-LIFETIME-WITH-AN-EXCEEDINGLY-LOW-PROBABILITY-OF GROSS-RUPTURE-FOR-ANY-CONDITION-OF-MATERIALS,-COMPONENT-PARTS,-OR-OPERATING ENVIRONMENT-THAT-MIGHT-REASONABLY-BE-POSTULATED.]

CRITERION 7 - CONTAINMENT (Category A)

[A-HIGH-INTEGRITY-CONTAINMENT-STRUCTURE-THAT-PROVIDES-A-PROTECTIVE-CAPABILITY-BEYOND-THE-CAPACITY-OF-THE-CORE-AND-COOLANT-BOUNDARY-FOR-ACCOMMODATING THE-ABNORMAL.] THE CONTAINMENT [SUCH-A] STRUCTURE SHALL BE DESIGNED TO SUSTAIN THE INITIAL EFFECTS OF GROSS EQUIPMENT FAILURES, SUCH AS A LARGE COOLANT BOUNDARY BREAK, WITHOUT LOSS OF REQUIRED INTEGRITY AND, TOGETHER WITH OTHER ENGINEERED SAFEGUARDS AS MAY BE NECESSARY, TO RETAIN FOR AS LONG AS THE SITUATION REQUIRES THE FUNCTIONAL CAPABILITY TO PROTECT THE PUBLIC.

III. NUCLEAR AND RADIATION PROCESS CONTROLS

CRITERION 8 - CONTROL ROOM (Category B)

THE FACILITY SHALL BE PROVIDED WITH A [DESIGNED-SO-THAT-OPERATIONS-CAN-BE MONITORED-AND-CONTROLLED-AT-ALL-TIMES.--THERE-SHALL-BE-INCLUDED] CONTROL ROOM FROM WHICH ACTIONS [CAN-BE-CONTROLLED] TO MAINTAIN SAFE OPERATIONAL STATUS OF THE PLANT CAN BE CONTROLLED. ADEQUATE RADIATION PROTECTION SHALL BE PROVIDED TO PERMIT ACCESS, EVEN UNDER ACCIDENT CONDITIONS, TO EQUIPMENT IN THE CONTROL

ROOM OR OTHER AREAS AS NECESSARY TO SHUT DOWN AND MAINTAIN SAFE CONTROL OF THE FACILITY WITHOUT RADIATION /UNDUE/ EXPOSURES OF PERSONNEL IN EXCESS /AS ESTABLISHED-BY-CONSIDERATIONS/ OF 10 CFR 20 LIMITS.

CRITERION 9 - PROCESS CONTROL SYSTEMS (Category B)

PROCESS INSTRUMENTATION AND CONTROLS SHALL BE PROVIDED AS REQUIRED TO MONITOR AND MAINTAIN PROCESS VARIABLES WITHIN NOMINAL OPERATING RANGES. /PROCESS-CONTROLS-AS-REQUIRED-TO-MONITOR-AND-MAINTAIN-PROCESS-VARIABLES-WITHIN NOMINAL-OPERATING-RANGES-AND-TO-PREVENT-OR-SUPPRESS-POWER-OR-PROCESS-VARIABLE OSCILLATIONS-OR-TRANSIENTS-THAT-COULD-RESULT-IN-EXCEEDING-FUEL-DAMAGE-LIMITS-AS ESTABLISHED-FOR-NORMAL-OPERATION. ALSO, SEE CRITERION 4./

CRITERION 10 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

MEANS SHALL BE PROVIDED FOR MONITORING AND MAINTAINING CONTROL OVER COM- PONENTS, PROCESSES, /THE-FISSION-PROCESS-THROUGHOUT-CORE-LIFE/ AND /FOR-ALL/ CONDITIONS THAT CAN REASONABLY BE ANTICIPATED TO CAUSE VARIATIONS IN REACTIVITY OF THE CORE; SUCH AS, INDICATION OF POSITION /INDICATION/ OF CONTROL RODS, /AND/ CONCENTRATION OF SOLUBLE REACTIVITY CONTROL POISONS, AND DISPOSITION OF FUEL.

CRITERION 11 - CORE PROTECTIVE SYSTEMS (Category B)

CORE PROTECTIVE SYSTEMS, TOGETHER WITH ASSOCIATED EQUIPMENT, SHALL BE DESIGNED TO ACT AUTOMATICALLY TO PREVENT OR SUPPRESS CONDITIONS THAT COULD RESULT IN EXCEEDING FUEL DAMAGE LIMITS. /PROTECTIVE-INSTRUMENTATION-DESIGNED TO-PREVENT,-BY-AUTOMATIC-ACTION-OF-REACTIVITY-CONTROLS,-EXCEEDING-DESIGN-LIMITS ESTABLISHED-FROM-CONSIDERATIONS-OF-FUEL-DAMAGE./

CRITERION 12 - ENGINEERED SAFEGUARDS PROTECTIVE SYSTEMS (Category B)

PROTECTIVE SYSTEMS /INSTRUMENTATION/ SHALL BE PROVIDED FOR SENSING ACCIDENT /ABNORMAL/ SITUATIONS AND INITIATING THE OPERATION OF NECESSARY ENGINEERED SAFEGUARDS. /OR-PERFORMING-PREREQUISITE-FUNCTIONS-SUCH-AS-VALVE-ACTUATION./

CRITERION 13 - MONITORING RADIOACTIVITY RELEASES (Category B)

MEANS SHALL BE PROVIDED FOR MONITORING THE CONTAINMENT ATMOSPHERE, THE FACILITY EFFLUENT DISCHARGE PATHS, AND THE FACILITY ENVIRONS FOR RADIOACTIVITY THAT MIGHT BE RELEASED EITHER AS A RESULT OF NORMAL OPERATIONS OR ACCIDENT CONDITIONS /ABNORMAL-SITUATIONS/.

CRITERION 14 - MONITORING FUEL AND WASTE STORAGE (Category B)

MONITORING AND ALARM INSTRUMENTATION SHALL BE PROVIDED FOR FUEL AND WASTE STORAGE AND HANDLING AREAS /PRINCIPALLY/ FOR CONDITIONS THAT MIGHT CONTRIBUTE /TO-INADVERTENT-CRITICALITY/ TO LOSS OF CONTINUITY IN DECAY HEAT REMOVAL AND TO RADIATION EXPOSURES.

IV. RELIABILITY AND TESTABILITY OF PROTECTIVE SYSTEMS

CRITERION 15 - PROTECTIVE SYSTEM RELIABILITY (Category B)

PROTECTIVE SYSTEMS SHALL BE DESIGNED FOR HIGH FUNCTIONAL RELIABILITY AND IN-SERVICE TESTABILITY /RELIABILITY-SHALL-BE/ COMMENSURATE WITH THE SAFETY FUNCTIONS TO BE PERFORMED.

CRITERION 16 - PROTECTIVE SYSTEM REDUNDANCY AND INDEPENDENCE (Category B)

REDUNDANCY AND INDEPENDENCE DESIGNED INTO PROTECTIVE SYSTEMS SHALL BE SUFFICIENT TO ASSURE THAT NO SINGLE FAILURE OR REMOVAL FROM SERVICE OF ANY

COMPONENT OR CHANNEL OF A SYSTEM WILL RESULT IN LOSS OF THE PROTECTIVE FUNCTION. THE REDUNDANCY PROVIDED SHALL INCLUDE, AS A MINIMUM, TWO CHANNELS OF PROTECTION FOR EACH PROTECTIVE FUNCTION TO BE SERVED.

CRITERION 17 - SINGLE FAILURE DEFINITION (Category B)

MULTIPLE FAILURES RESULTING FROM A SINGLE EVENT SHALL BE TREATED AS A SINGLE FAILURE.

CRITERION 18 - SEPARATION OF PROTECTIVE AND PROCESS CONTROL SYSTEMS (Category B)

PROTECTIVE SYSTEMS SHALL BE SEPARATED FROM PROCESS CONTROL SYSTEMS TO THE EXTENT THAT FAILURE OF REMOVAL FROM SERVICE OF ANY PROCESS CONTROL SYSTEM COMPONENT OR CHANNEL, OR THOSE COMMON TO PROCESS CONTROL AND PROTECTIVE CIRCUITRY, SHALL NOT NEGATE THE MINIMUM REDUNDANCY REQUIREMENTS FOR THE PROTECTIVE CHANNELS.

CRITERION 19 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTIVE SYSTEMS (Category B)

THE EFFECTS OF ADVERSE CONDITIONS TO WHICH REDUNDANT CHANNELS OF PROTECTIVE SYSTEMS MIGHT BE EXPOSED IN COMMON, EITHER UNDER NORMAL CONDITIONS OR THOSE OF AN ACCIDENT, SHALL NOT RESULT IN LOSS OF THE PROTECTIVE FUNCTION.

CRITERION 20 - EMERGENCY POWER FOR PROTECTIVE SYSTEMS (Category B)

IN THE EVENT OF LOSS OF ALL OFFSITE POWER, SUFFICIENT SOURCES OF /POWER/ ALTERNATE POWER /~~TO-THE-NORMAL-SUPPLY~~/ SHALL BE PROVIDED TO ASSURE A CAPABILITY FOR PERFORMING THE /~~PROTECTIVE~~/ FUNCTIONS OF THE PROTECTIVE SYSTEMS.

CRITERION 21 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTIVE SYSTEMS (Category B)

MEANS SHALL BE INCLUDED FOR TESTING PROTECTIVE SYSTEMS /~~INSTRUMENTATION~~/

WHILE THE REACTOR IS IN OPERATION TO DEMONSTRATE FUNCTIONAL OPERABILITY AND TO DETERMINE COMPONENT OR CIRCUIT FAILURES

CRITERION 22 - PROTECTIVE SYSTEMS FAIL-SAFE DESIGN (Category B)

THE PROTECTIVE SYSTEMS SHALL BE DESIGNED [A-LOGIC-THAT-MAKES-THE-INSTRUMENTATION] TO GO INTO A SAFE STATE OR A STATE ESTABLISHED AS TOLERABLE ON SOME OTHER BASIS IF CONDITIONS SUCH AS GROSS DISCONNECTION OF THE SYSTEM, LOSS OF ENERGY (ELECTRIC POWER, INSTRUMENT AIR) OR ADVERSE ENVIRONMENTS (EXTREME HEAT OR COLD, FIRE, STEAM, OR WATER) ARE EXPERIENCED.

CRITERION 23 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

AT LEAST TWO INDEPENDENT REACTIVITY CONTROL SYSTEMS, PREFERABLY OF DIFFERENT PRINCIPLES, SHALL BE PROVIDED. [A-SECONDARY-OR-BACKUP-REACTIVITY-CONTROL-MEANS INDEPENDENT-OF-THE-PRIMARY-METHOD-OF-REACTIVITY-SHUTDOWN-WITH-CAPABILITY-TO-SHUT DOWN-THE-REACTOR-FROM-ANY-OPERATING-CONDITION.]

CRITERION 24 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

AT LEAST TWO OF THE REACTIVITY CONTROL SYSTEMS PROVIDED SHALL INDEPENDENTLY BE CAPABLE OF MAKING AND HOLDING THE CORE SUBCRITICAL AT ANY HOT STANDBY OR HOT OPERATING CONDITION SUFFICIENTLY FAST TO PREVENT EXCEEDING FUEL DAMAGE LIMITS.

V. REACTIVITY CONTROL

CRITERION 25 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

AT LEAST ONE OF THE REACTIVITY CONTROL SYSTEMS PROVIDED SHALL BE CAPABLE OF MAKING THE CORE SUBCRITICAL UNDER ANY REACTOR CONDITION (INCLUDING TRANSIENTS)

SUFFICIENTLY FAST TO PREVENT EXCEEDING FUEL DAMAGE LIMITS. SHUTDOWN MARGINS GREATER THAN THE MAXIMUM WORTH OF THE MOST EFFECTIVE CONTROL ROD WHEN FULLY WITHDRAWN SHALL BE PROVIDED.

CRITERION 26 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

AT LEAST ONE OF THE REACTIVITY CONTROL SYSTEMS PROVIDED SHALL BE CAPABLE OF MAKING AND HOLDING THE /REACTIVITY-CONTROL-SHALL-INCLUDE-A-SHUTDOWN-CAPABILITY-SUFFICIENT-TO-MAKE-AND-HOLD/ CORE SUBCRITICAL UNDER ANY CONDITIONS /FROM-ANY-OPERATING/ WITH APPROPRIATE MARGINS FOR CONTINGENCIES.

CRITERION 27 - REACTIVITY CONTROL SYSTEM MALFUNCTION (Category B)

THE REACTIVITY CONTROL SYSTEMS SHALL BE CAPABLE OF SUSTAINING /REACTIVITY-CONTROL-SHALL-INCLUDE-A-CAPABILITY-TO-SUSTAIN/ ANY SINGLE /CONTROL-SYSTEM/ MALFUNCTION WITHOUT CAUSING A REACTIVITY TRANSIENT WHICH COULD RESULT /RESULTS/ IN EXCEEDING FUEL DAMAGE LIMITS /CORE-DESIGN-LIMITS-ESTABLISHED-FROM-CONSIDERATIONS-OF-FUEL-DAMAGE/.

CRITERION 28 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

LIMITS SHALL BE PLACED ON /UPON/ MAXIMUM REACTIVITY WORTH OF CONTROL RODS OR ELEMENTS AND /THE/ RATES AT WHICH REACTIVITY CAN BE INCREASED TO ENSURE THAT THE POTENTIAL EFFECTS OF A SUDDEN OR LARGE CHANGE OF REACTIVITY CANNOT (a) RUPTURE THE PRIMARY PRESSURE BOUNDARY OR (b) DISRUPT THE CORE, ITS SUPPORT STRUCTURES, OR OTHER VESSEL INTERNALS SUFFICIENTLY TO IMPAIR THE EFFECTIVENESS OF EMERGENCY CORE COOLING /THEY-MIGHT-BE-INSERTED-UNDER-DESIGN-LIMITING-SITUATIONS,-SUCH-AS A-SUDDEN-REACTIVITY-INSERTION-OR-LOSS-OF-COOLANT,-TO-ASSURE-THE-CAPABILITY-FOR POSTACCIDENT-SHUTDOWN-OF-THE-REACTOR-AND-THE-AVOIDANCE-OF-POTENTIAL-EFFECTS-THAT COULD-DISRUPT-THE-INTERNALS-OF-THE-REACTOR-OR-BREAK-THE-PRIMARY-PRESSURE-BOUNDARY/

VI. REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 29 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

/~~THE PRIMARY COOLANT PRESSURE BOUNDARY SHALL BE DESIGNED ON THE BASIS THAT:~~/ THE REACTOR COOLANT PRESSURE BOUNDARY SHALL /~~MUST~~/ BE CAPABLE OF ACCOMMODATING WITHOUT RUPTURE, AND WITH, AT MOST, LIMITED NEED FOR ENERGY ABSORPTION THROUGH PLASTIC DEFORMATION, THE STATIC AND DYNAMIC LOADS IMPOSED ON ANY BOUNDARY COMPONENT AS A RESULT OF AN INADVERTENT AND SUDDEN RELEASE OF ENERGY TO THE COOLANT. AS A DESIGN REFERENCE, THIS SUDDEN RELEASE SHALL BE TAKEN AS THAT WHICH WOULD RESULT FROM A SUDDEN REACTIVITY INSERTION SUCH AS /~~A~~/ ROD EJECTION /~~COMMONLY POSTULATED FOR PWRs~~/ AND /~~A~~/ ROD DROPOUT ACCIDENTS /~~FOR THE BWRs~~/.

CRITERION 30 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

THE REACTOR COOLANT PRESSURE BOUNDARY SHALL BE DESIGNED SO THAT RAPID PROPAGATION TYPE FAILURES ARE PRECLUDED. DUE CONSIDERATION SHALL BE GIVEN TO THE NOTCH-TOUGHNESS PROPERTIES OF MATERIALS, THE STATE OF STRESS UNDER STATIC AND TRANSIENT LOADINGS, THE QUALITY CONTROL SPECIFIED FOR MATERIALS AND COMPONENT FABRICATION TO LIMIT FLAW SIZES, AND THE PROVISIONS FOR CONTROL OVER SERVICE TEMPERATURES AND IRRADIATION EFFECTS.

CRITERION 31 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

UNDER CONDITIONS WHERE REACTOR COOLANT PRESSURE BOUNDARY SYSTEM COMPONENTS CONSTRUCTED OF FERRITIC MATERIALS MAY BE SUBJECTED TO POTENTIAL REACTIVITY-INDUCED LOADINGS, SERVICE TEMPERATURES SHALL BE AT LEAST 120°F ABOVE THE NDT TEMPERATURE

OF THE COMPONENT MATERIAL IF THE RESULTING ENERGY RELEASE IS EXPECTED TO BE ABSORBED BY PLASTIC DEFORMATION OR 60°F ABOVE THE NDT TEMPERATURE OF THE COMPONENT MATERIAL IF THE RESULTING ENERGY RELEASE IS EXPECTED TO BE ABSORBED WITHIN THE ELASTIC STRAIN ENERGY RANGE. /~~COMPONENTS IN THE COOLANT SYSTEM WHICH ARE POTENTIALLY SUBJECT TO PROPAGATION TYPE OF FAILURES WILL NOT BE ALLOWED TO HAVE SUBSTANTIAL PRESSURE OR THERMAL STRESS IMPOSED WHILE TEMPERATURES ARE BELOW VALUES FOR WHICH DUCTILE BEHAVIOR OF THE MATERIALS CANNOT BE ASSURED. MORE SPECIFICALLY, FOR FERRITIC MATERIALS SUCH AS 302B AND A212 COMMONLY EMPLOYED IN PRESSURE VESSELS, THE FOLLOWING APPLY:~~

~~WHERE ENERGY ABSORPTION BY PLASTIC DEFORMATION IS NECESSARY TO MEET THE REQUIREMENTS OF PREVENTING RUPTURE OF THE PRIMARY PRESSURE BARRIER UNDER REACTIVITY INDUCED LOADINGS, THE REACTOR SHALL BE DESIGNED TO OPERATE AT OR ABOVE THE REACTOR VESSEL FAILURE TRANSITION PLASTIC TEMPERATURE (FTP).~~

~~WHERE ENERGY ABSORPTION BY PLASTIC DEFORMATION IS NOT REQUIRED TO ACCOMMODATE REACTIVITY INDUCED LOADINGS, THE REACTOR SHALL BE DESIGNED FOR OPERATION AND TESTING AT OR ABOVE THE FAILURE TRANSITION ELASTIC TEMPERATURE (FTE).~~/

CRITERION 32 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS SHALL HAVE PROVISIONS FOR INSPECTION, TESTING, AND SURVEILLANCE BY APPROPRIATE MEANS TO ASSESS THE STRUCTURAL AND LEAKTIGHT INTEGRITY OF THE BOUNDARY COMPONENTS DURING ITS SERVICE LIFETIME. FOR THE REACTOR VESSEL, A MATERIAL SURVEILLANCE PROGRAM CONFORMING WITH ASTM-E-185-66 SHALL BE PROVIDED.

VII. ENGINEERED SAFEGUARDS

CRITERION 33 - ENGINEERED SAFEGUARDS BASIS FOR DESIGN (Category A)

SAFEGUARDS SHALL BE PROVIDED [ENGINEERED] IN [INTO] THE FACILITY TO BACK UP SAFETY FEATURES PROVIDED BY THE CORE DESIGN AND THE CORE AND COOLANT BOUNDARY PROTECTIVE SYSTEMS. AS A MINIMUM, SUCH SAFEGUARDS SHALL BE DESIGNED [ON-THE-BASIS-THAT,--INTEGRITY-OF-THE-COOLANT-BOUNDARY-IS-LOST.--AS-A-MINIMUM,-THE-DESIGN-SHALL] TO ACCOMMODATE A RANGE OF PRIMARY COOLANT SYSTEM BREAKS UP TO AND INCLUDING THE CIRCUMFERENTIAL RUPTURE OF ANY PIPE IN THAT [THE-REACTOR-COOLANT] SYSTEM ASSUMING UNOBSTRUCTED DISCHARGE FROM BOTH ENDS.

CRITERION 34 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFEGUARDS (Category A)

ALL ENGINEERED SAFEGUARDS SHALL BE DESIGNED TO PROVIDE HIGH FUNCTIONAL RELIABILITY AND READY TESTABILITY. IN DETERMINING THE SUITABILITY OF A FACILITY FOR A PROPOSED SITE, THE DEGREE OF PERMISSIBLE RELIANCE UPON AND ACCEPTANCE OF THE INHERENT AND ENGINEERED SAFETY AFFORDED BY THE SYSTEMS, MATERIALS AND COMPONENTS, AND THE ASSOCIATED ENGINEERED SAFETY AFFORDED BY THE SYSTEMS, MATERIALS AND COMPONENTS, AND THE ASSOCIATED ENGINEERED SAFEGUARDS TO BE BUILT INTO THE FACILITY WILL BE INFLUENCED BY THEIR KNOWN OR THEIR DEMONSTRATED PERFORMANCE CAPABILITY AND RELIABILITY AND THE EXTENT TO WHICH THE OPERABILITY OF SUCH SYSTEMS, MATERIALS, COMPONENTS, AND ENGINEERED SAFEGUARDS CAN BE TESTED AND INSPECTED WHERE APPROPRIATE DURING THE LIFE OF THE PLANT. [THEREFORE,-ALL-ENGINEERED-SAFEGUARDS-SHALL-BE-DESIGNED-TO-PROVIDE-HIGH-FUNCTIONAL-RELIABILITY-AND-READY-TESTABILITY.]

CRITERION 35 - EMERGENCY POWER FOR ENGINEERED SAFEGUARDS (Category A)

IN THE EVENT OF LOSS OF ALL OFFSITE POWER, [THERE-MUST-BE-PROVIDED] SUFFICIENT SOURCES OF [POWER] ALTERNATE POWER SHALL BE PROVIDED [TO-THE-NORMAL-SUPPLY] TO ASSURE A CAPABILITY FOR PERFORMING THE [ALL] FUNCTIONS REQUIRED OF THE ENGINEERED SAFEGUARDS [FOR-PUBLIC-SAFETY-UNDER-ALL-CREDIBLE-CIRCUMSTANCES].

CRITERION 36 - MISSILE PROTECTION (Category A)

PROTECTION FOR ENGINEERED SAFEGUARDS SHOULD BE PROVIDED [IS-REQUIRED] FROM DYNAMIC EFFECTS AND MISSILES THAT MIGHT RESULT FROM PLANT EQUIPMENT FAILURES.

CRITERION 37 - ENGINEERED SAFEGUARD SYSTEM PERFORMANCE CAPABILITY (Category A)

ENGINEERED SAFEGUARD [REDUNDANT] SYSTEMS SUCH AS EMERGENCY CORE COOLING AND CONTAINMENT HEAT REMOVAL SHALL PROVIDE SUFFICIENT RELIABILITY AND PERFORMANCE CAPABILITY TO ACCOMMODATE PARTIAL LOSS OF INSTALLED CAPACITY AND STILL FULFILL THE REQUIRED SAFETY FUNCTION. [REDUNDANT-COMPONENTS-AND-SYSTEMS-WHERE-EMPLOYED SHALL-BE-INDEPENDENT-ONE-FROM-ANOTHER.]

CRITERION 38 - ACCIDENT AGGRAVATION PREVENTION (Category A)

ENGINEERED SAFEGUARDS SHALL BE DESIGNED SO THAT ACTION OF THE SAFEGUARDS WHICH [THAT] MIGHT ACCENTUATE THE ADVERSE AFTER-EFFECTS OF THE LOSS OF NORMAL COOLING BY EITHER PLANNED OR INADVERTENT OPERATION OF THE ENGINEERED SAFEGUARDS IS [TO-BE] AVOIDED.

CRITERION 39 - EMERGENCY CORE COOLING (Category A)

THE EMERGENCY CORE COOLING SYSTEM SHALL BE DESIGNED TO PREVENT FUEL AND CLAD DAMAGE THAT WOULD INTERFERE WITH ADEQUATE EMERGENCY CORE COOLING AND TO LIMIT THE CLAD METAL-WATER REACTION TO NEGLIGIBLE AMOUNTS FOR ALL SIZES OF

BREAKS IN THE REACTOR COOLANT PIPING UP TO AND INCLUDING THE DOUBLE-ENDED RUPTURE OF THE LARGEST PIPE. [THE-EFFECTS-OF-FUEL-TEMPERATURES-FROM-DECAY-HEAT-AND-FROM CHEMICAL-REACTIONS-THAT-COULD-CAUSE-OR-MATERIALLY-AUGMENT-THE-RELEASE-OF-FISSION PRODUCTS-FROM-THE-CORE-ARE-TO-BE-LIMITED-BOTH-IN-EXTENT-AND-CONSEQUENCES.--TEM-PERATURES-THAT-COULD-ENDANGER-THE-CAPABILITY-OF-THE-REACTOR-VESSEL-TO-FUNCTION AS-A-POSTACCIDENT-CORE-ENCLOSURE-AND-COOLANT-CONTAINER-ARE-TO-BE-PREVENTED.]

CRITERION 40 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

DESIGN PROVISIONS SHALL BE MADE TO FACILITATE [EMERGENCY-CORE-COOLING SYSTEM-SHALL-BE-DESIGNED-SO-THAT:] PHYSICAL INSPECTION OF ALL CRITICAL PARTS OF THE EMERGENCY COOLING SYSTEMS [COMPONENTS] INCLUDING REACTOR VESSEL INTERNALS AND WATER INJECTION NOZZLES [IN-CLOSED-LOOP-PIPING,-CAN-BE-ACCOMPLISHED].

CRITERION 41 - TESTING OF EMERGENCY CORE COOLING COMPONENTS (Category A)

DESIGN PROVISIONS SHALL BE MADE [EMERGENCY-CORE-COOLING-SYSTEM-SHALL-BE DESIGNED] SO THAT ACTIVE COMPONENTS OF THE EMERGENCY CORE COOLING SYSTEMS, SUCH AS PUMPS AND VALVES, CAN BE TESTED PERIODICALLY FOR OPERABILITY AND REQUIRED FUNCTIONAL PERFORMANCE.

CRITERION 42 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

[EMERGENCY-CORE-COOLING-SYSTEM-SHALL-BE-DESIGNED-SO-THAT:] A CAPABILITY SHALL BE [IS] PROVIDED TO TEST PERIODICALLY THE DELIVERY CAPABILITY OF THE EMERGENCY CORE COOLING SYSTEMS AT A LOCATION [POSITION] AS CLOSE TO THE CORE AS IS PRACTICAL.

CRITERION 43 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A CAPABILITY SHALL BE [IS] PROVIDED TO TEST UNDER CONDITIONS AS CLOSE TO DESIGN AS PRACTICAL THE FULL OPERATIONAL SEQUENCE THAT WOULD BRING THE EMERGENCY CORE COOLING SYSTEMS INTO ACTION, INCLUDING THE TRANSFER TO ALTERNATE POWER SOURCES.

CRITERION 44 - CONTAINMENT DESIGN BASIS (Category A)

THE CONTAINMENT STRUCTURE, INCLUDING ACCESS OPENINGS AND PENETRATIONS, SHALL BE DESIGNED TO ACCOMMODATE OR DISSIPATE WITHOUT EXCEEDING THE DESIGN LEAKAGE RATE THE LARGEST CREDIBLE ENERGY RELEASE, INCLUDING THE EFFECTS OF CREDIBLE METAL-WATER OR OTHER CHEMICAL REACTIONS THAT COULD OCCUR IN THE ABSENCE OF EMERGENCY CORE COOLING [CORE-QUENCHING] SYSTEMS.

CRITERION 45 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

PRINCIPAL LOAD CARRYING COMPONENTS OF FERRITIC MATERIALS EXPOSED TO THE EXTERNAL ENVIRONMENT SHALL BE SELECTED SO THAT THEIR TEMPERATURES UNDER NORMAL OPERATING AND TESTING CONDITIONS ARE NOT LESS THAN 30°F ABOVE NIL-DUCTILITY TRANSITION (NDT) TEMPERATURE. [WELL-ABOVE-NDT---THE-MARGINS-ABOVE-CONSERVATIVELY ESTIMATED-NIL-DUCTILITY-TRANSITION-TEMPERATURES-IN-NO-CASE-SHALL-BE-LESS-THAN 30°F.]

CRITERION 46 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

IF PART OF THE REACTOR [PRIMARY] COOLANT PRESSURE BOUNDARY IS OUTSIDE THE CONTAINMENT, APPROPRIATE SAFEGUARDS AS NECESSARY SHALL BE PROVIDED TO PROTECT THE HEALTH AND SAFETY OF THE PUBLIC IN CASE OF AN ACCIDENTAL RUPTURE IN THAT

PART. DETERMINATION OF THE APPROPRIATENESS OF SAFEGUARDS SUCH AS ISOLATION VALVES AND ADDITIONAL CONTAINMENT /ETC./ SHALL INCLUDE CONSIDERATION OF THE ENVIRONMENTAL AND POPULATION CONDITIONS SURROUNDING THE SITE.

CRITERION 47 - CONTAINMENT HEAT REMOVAL (Category A)

WHERE ACTIVE HEAT REMOVAL /DISSIPATION/ SYSTEMS ARE NEEDED UNDER ACCIDENT CONDITIONS TO PREVENT EXCEEDING CONTAINMENT DESIGN PRESSURE, AT LEAST TWO SYSTEMS SHALL BE PROVIDED, PREFERABLY OF DIFFERENT PRINCIPLES.

CRITERION 48 - CONTAINMENT ISOLATION VALVES (Category A)

PENETRATIONS THAT REQUIRE CLOSURE FOR THE CONTAINMENT FUNCTIONS SHALL BE /MUST-BE/ PROTECTED BY MULTIPLE VALVING AND ASSOCIATED APPARATUS.

CRITERION 49 - CONTAINMENT LEAK TEST (Category A)

CONTAINMENT SHALL BE DESIGNED SO THAT AN INTEGRATED LEAK TEST CAN BE CONDUCTED AT LEAST TO DESIGN PRESSURE AFTER COMPLETION AND INSTALLATION OF ALL PENETRATIONS AND THE LEAKAGE RATE MEASURED OVER A SUFFICIENT PERIOD OF TIME TO VERIFY ITS CONFORMANCE WITH REQUIRED PERFORMANCE.

CRITERION 50 - CONTAINMENT PERIODIC TESTING (Category A)

THE CONTAINMENT SHALL BE DESIGNED SO THAT INTEGRATED LEAKAGE RATE TESTING CAN BE DONE AT DESIGN PRESSURE /PERIODICALLY/ DURING PLANT LIFETIME

CRITERION 51 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

PROVISIONS SHALL BE MADE FOR TESTING ALL PENETRATIONS SUBJECT TO FAILURE OR DETERIORATION IN SERVICE SUCH AS RESILIENT SEALS AND EXPANSION BELLOWS TO PERMIT LEAKTIGHTNESS TO BE DEMONSTRATED AT DESIGN PRESSURE AT ANY TIME.

CRITERION 52 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

CAPABILITY SHALL BE /IS/ PROVIDED FOR TESTING FUNCTIONAL OPERABILITY OF VALVES AND ASSOCIATED APPARATUS ESSENTIAL TO THE CONTAINMENT FUNCTION FOR ESTABLISHING WHETHER /THAT-NO/ FAILURE HAS OCCURRED AND FOR DETERMINING THAT VALVE LEAKAGE DOES NOT EXCEED ACCEPTABLE LIMITS.

CRITERION 53 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

DESIGN PROVISIONS SHALL BE MADE TO FACILITATE THE PERIODIC PHYSICAL INSPECTION OF ALL IMPORTANT COMPONENTS OF THE CONTAINMENT PRESSURE-REDUCING SYSTEMS; SUCH AS, PUMPS, VALVES, SPRAY NOZZLES, AND SUMPS. /CONTAINMENT-PRESSURE-REDUCING-SYSTEMS-SHALL-BE-DESIGNED-SO-THAT-PHYSICAL-INSPECTION-OF-ALL-COMPONENTS,-SUCH-AS,-SPRAY-NOZZLES-AND-SUMPS,-CAN-BE-ACCOMPLISHED./

CRITERION 54 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEM COMPONENTS (Category A)

THE CONTAINMENT PRESSURE-REDUCING SYSTEMS SHALL BE DESIGNED SO THAT ACTIVE COMPONENTS, SUCH AS PUMPS AND VALVES, CAN BE TESTED PERIODICALLY FOR OPERABILITY AND REQUIRED FUNCTIONAL PERFORMANCE.

CRITERION 55 - TESTING OF CONTAINMENT SPRAY SYSTEM (Category A)

A CAPABILITY SHALL BE /IS/ PROVIDED TO TEST PERIODICALLY THE DELIVERY CAPABILITY OF THE CONTAINMENT SPRAY SYSTEM AT A POSITION AS CLOSE TO THE SPRAY NOZZLES AS IS PRACTICAL.

CRITERION 56 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A CAPABILITY SHALL BE /IS/ PROVIDED TO TEST UNDER CONDITIONS AS CLOSE TO THE DESIGN AS PRACTICAL THE FULL OPERATIONAL SEQUENCE THAT WOULD BRING THE

CONTAINMENT PRESSURE-REDUCING SYSTEMS INTO ACTION, INCLUDING THE TRANSFER TO ALTERNATE POWER SOURCES.

CRITERION 57 - INSPECTION OF AIR CLEANUP COMPONENTS (Category A)

DESIGN PROVISIONS SHALL BE MADE TO FACILITATE PHYSICAL INSPECTION OF ALL CRITICAL PARTS OF CONTAINMENT AIR CLEANUP SYSTEMS; SUCH AS, DUCTS, FILTERS, FANS, AND DAMPERS. [CONTAINMENT-AIR-CLEANUP-SYSTEMS-SHALL-BE-DESIGNED-SO-THAT PHYSICAL-INSPECTION-OF-ALL-COMPONENTS,-INCLUDING-DUCT-WORK-AND-FILTER-INSTALLA-TION,-CAN-BE-ACCOMPLISHED.]

CRITERION 58 - TESTING OF AIR CLEANUP SYSTEMS ACTIVE COMPONENTS (Category A)

DESIGN PROVISIONS SHALL BE MADE SO THAT ACTIVE COMPONENTS OF THE [CONTAINMENT] AIR CLEANUP SYSTEMS, SUCH AS FANS AND DAMPERS, CAN BE TESTED PERIODICALLY FOR OPERABILITY AND REQUIRED FUNCTIONAL PERFORMANCE.

CRITERION 59 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A CAPABILITY SHALL BE [IS] PROVIDED FOR IN SITU PERIODIC TESTING AND SURVEILLANCE OF THE [CONTAINMENT] AIR CLEANUP SYSTEMS TO ENSURE (a) FILTER BYPASS PATHS HAVE NOT DEVELOPED AND (b) FILTER AND TRAPPING MATERIALS HAVE NOT DETERIORATED BEYOND ACCEPTABLE LIMITS.

CRITERION 60 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A CAPABILITY SHALL BE [IS] PROVIDED TO TEST UNDER CONDITIONS AS CLOSE TO DESIGN AS PRACTICAL THE FULL OPERATIONAL SEQUENCE THAT WOULD BRING THE AIR CLEANUP SYSTEM INTO ACTION, INCLUDING THE TRANSFER TO ALTERNATE POWER SOURCES AND THE DESIGN AIR FLOW DELIVERY CAPABILITY.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 61 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

POSSIBILITIES FOR /INADVERTENT/ CRITICALITY IN NEW AND SPENT FUEL STORAGE SHALL BE /MUST-BE/ PREVENTED BY PHYSICAL /ENGINEERED/ SYSTEMS OR PROCESSES TO EVERY EXTENT PRACTICABLE. SUCH MEANS AS FAVORABLE GEOMETRIES /GEOMETRIC-SAFE-SPACING-LIMITS/ SHALL BE EMPHASIZED OVER PROCEDURAL CONTROLS.

CRITERION 62 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

RELIABLE DECAY HEAT REMOVAL SYSTEMS SHALL BE DESIGNED TO ENSURE DAMAGE TO THE FUEL OR STORAGE FACILITIES /MEANS-MUST-BE-PROVIDED-AS-NECESSARY-TO PREVENT-FUEL-OR-STORAGE-VOLUME-DAMAGE/ THAT COULD RESULT IN RADIOACTIVITY RELEASE TO PLANT OPERATING AREAS OR THE PUBLIC ENVIRONS IS PREVENTED. SUCH MEANS MUST BE ASSURED FOR ALL ANTICIPATED NORMAL AND ABNORMAL CONDITIONS AS WELL AS THOSE ACCIDENT SITUATIONS WHEREBY NORMAL COOLING COULD CREDIBLY BECOME LOST.

CRITERION 63 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category A)

SHIELDING FOR RADIATION PROTECTION SHALL BE PROVIDED IN THE DESIGN OF SPENT FUEL AND WASTE STORAGE FACILITIES AS REQUIRED FROM CONSIDERATION/S/ OF 10 CFR 20.

CRITERION 64 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

CONTAINMENT OF FUEL AND WASTE STORAGE /THE-SYSTEMS/ SHALL BE PROVIDED IF ACCIDENTS COULD LEAD TO RELEASE OF UNDUE AMOUNTS OF RADIOACTIVITY TO THE PUBLIC ENVIRONS.

IX. PLANT EFFLUENTS

CRITERION 65 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

THE FACILITY DESIGN SHALL INCLUDE THOSE MEANS NECESSARY TO MAINTAIN CONTROL OVER PLANT RADIOACTIVE EFFLUENTS, WHETHER SOLID, LIQUID, OR GASEOUS. APPROPRIATE HOLDUP CAPACITY SHALL BE PROVIDED FOR RETENTION OF GASEOUS LIQUID, OR SOLID EFFLUENTS, PARTICULARLY WHERE UNFAVORABLE ENVIRONMENTAL CONDITIONS CAN BE EXPECTED TO REQUIRE OPERATIONAL LIMITATIONS UPON THE RELEASE OF RADIOACTIVE EFFLUENTS TO THE ENVIRONMENT. IN ALL CASES, THE DESIGN FOR RADIOACTIVITY CONTROL SHALL BE JUSTIFIABLE ON THE BASIS OF (a) CONSIDERATIONS SET FORTH IN 10 CFR 20 FOR ANY NORMAL OPERATION OR ANY TRANSIENT SITUATION THAT MIGHT REASONABLY BE ANTICIPATED TO OCCUR OR (b) CONSIDERATIONS SET FORTH IN 10 CFR 100 FOR THE EXCEEDINGLY LOW PROBABILITY TYPE OF SITUATION AS DEPICTED THEREIN.

OUT

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 20, 1967

MEMORANDUM

To : H. E. Etherington, Design Criteria Subcommittee Chairman

From : S. H. Hanauer

Subject: REVIEW OF NEW DRAFT GENERAL DESIGN CRITERIA

Refer ·· (1) DiNunno letter 8 Feb. transmitting draft.
DiNunno letter 14 Feb. transmitting comparison of drafts.

Gene

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CRI-8 Reactor Design and Operating Criteria
1967-68

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3.b. I no longer understand what is ... rgins."

Chap 9
FN 13

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

February 20, 1967

MEMORANDUM

To : H. E. Etherington, Design Criteria Subcommittee Chairman

From : S. H. Hanauer

Subject: REVIEW OF NEW DRAFT GENERAL DESIGN CRITERIA

References: (1) DiNunno letter 8 Feb. transmitting draft.
(2) DiNunno letter 14 Feb. transmitting comparison of drafts.

General

These criteria are no worse and in some cases rather better than the July and October versions. They are still mostly harmless pious platitudes. Considering how the Staff has interpreted "fulfillment in principle" of some of the July criteria by some applicants, I see no quantum jump in safety from their promulgation. Neither do I see any reduction in ACRS workload from application of these admittedly incomplete criteria.

On the other hand, I see no great harm in promulgating these criteria.

Throughout the document, "protective" should be replaced by "protection" as preferred image; see the recent IEEE criteria. "Protective" means a coating.

Specific (By criteria number of February 14 document).

1. I agree completely with comment 1a of Shaw's letter. Nowhere do these criteria call for the most necessary separation of vital from non vital equipment.

3.a. In view of recent discussions, I wonder if maybe the older drafts might have been correct in putting this in Category A. We are licensing a plant for a certain power. If at the construction permit stage we don't think that power is safe (under this criterion) we shouldn't say so. Hence Category A.

3.b. I no longer understand what is meant by "margins."

4. "Reliable process controls" could mean "adequate detection, plus a good administrative procedure for coping," which has been approved repeatedly. Unfortunately, I take "reliable process controls" to mean, "instrumentation and controls built to protection-system quality and reliability;" this is not meant, I think. The words "together with reliable process controls" should be deleted.

4.b. I do not know what "process oscillations" means.

5. Delete "together with its cooling and moderating systems." Moderation is part of the "core;" the cooling system does not enter. Taken literally, this clause allows an intrinsic positive power coefficient if some haywire coolant temperature controller makes an "apparent" negative coefficient.

6. Where did the exceedingly small rupture probability go? It will be missed. See Criterion 7, which discusses rupture consequences.

8. The old way was better, since it implied the necessity for retaining control if the control room becomes uninhabitable.

9. Delete the first "process."

10. This is now nonsensical. What are "means . . . for monitoring and maintaining control over . . . disposition of fuel?" Is this an instrumentation clause or a fuel hold-down clause or something else?

11. Why is "core" needed?

13. This criterion still does not contain the thought that the instrument range must be extended to indicate large, large releases that the designers think incredible.

17. This is really part of 16.

18.a. Delete "process" everywhere.

b. Change "shall not negate the minimum redundancy" (double negative) to "leaves intact a system satisfying all."

21. Change "functional operability" to "that no failures or loss of redundancy have occurred." Delete "and to determine component or circuit failures."

24.a. "Subcritical" at "hot operating!" Nonsense.

b. We do not enforce this on GE; their liquid poison system is nominal only.

25. Even under loss-of-coolant transients?

26. How does this differ from the last sentence of 25?

27. We do not now enforce this for the rod-drop (GE) or rod-ejection (PWR) accidents.

28. This is inadequate; there should be lots of margin in this one.

32. Is this enough, even for now?

35. The old 9.1.2 was better. The alternate power sources must be provided before all off-site power is lost. "A capability" is not enough; where is redundancy, testing, capacity margin, etc. etc.?

37. Delete "reliability and." This criterion is part of reliability. Why is the very important last sentence deleted?

38. Now inacceptable. The design must be such that the safeguards can work any time they get ready and not make things worse. The revision implies that interlocks should prevent protective (engineering safeguard) action at the wrong time — a dangerous kind of safeguard indeed.

39. Where did the margin go? See Palisades criteria. Unacceptable without margin.

52. Change "whether" back to "that no." The point (often repeated, and many times reinstated to this document) is that a valve can fail and yet a test can show no leakage because the backup valve is tight. Tests must reveal the first flaw so it can be fixed, else redundancy doesn't pay.

53. And toruses (tori?).

61. "Favorable" geometries are those favorable to criticality! Geometrically safe configurations" is the correct term.

62. Too many words between "ensure damage"(!) and "is prevented." Awful. Reword.

66. Sure is a lot of work making comments this way.

cc: S. H. Hanauer
F. A. Gifford
H. O. Monson
H. G. Mangelsdorf
D. Okrent
W. K. Ergen
N. J. Palladino

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June 16, 1967

ATOMIC ENERGY COMMISSION

PROPOSED AMENDMENT TO 10 CFR 50: GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Note by the Secretary

1. The Director of Regulation has requested that the attached report be circulated for consideration by the Commission at an early date.

2. The Commission approved the proposed design criteria, as revised, during consideration of AEC-R 2/49 at Regulatory Meeting 223 on November 10, 1965.

W. B. McCool
Secretary

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ATOMIC ENERGY COMMISSION

PROPOSED AMENDMENT TO 10 CFR 50: GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Report to the Director of Regulation by the Director, Division of Reactor Standards

THE PROBLEM

1. To consider the publication for public comment of a proposed amendment to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria for nuclear power plants to be included in applications for construction permits. Under the proposed amendments to this Part, specifically to §50.34, which were published for public comment in the Federal Register on August 16, 1966, applicants for an AEC construction permit would be required to specify these principal design criteria for a proposed facility. The proposed new guide would be substituted for the present Appendix A to Part 50.

BACKGROUND AND SUMMARY

2. The development and publication of criteria for nuclear power plants was one of the key recommendations of the Regulatory Review Panel which studied ways of streamlining the Commission's reactor licensing procedures. The Panel particularly stressed the need for design criteria to be used at the construction permit stage of a licensing proceeding. Work on the development of general criteria had been in progress at the time of the Review Panel's study. This effort was accelerated and led to the issuance in a Commission press release dated November 22, 1965, of draft criteria for use in the evaluation of applications for nuclear power plant construction permits.* The criteria were largely statements of design principles and objectives previously used by the staff in evaluating applications for reactor construction permits. Although they reflected the predominating experience with water reactors, they were considered to be generally applicable to other reactors as well.

*Secretariat Note: A copy of AEC press release H-252, November 22, 1965, is on file in the Office of the Secretary.

[illegible] groups of individuals submitted comments, as listed in Appendix "A." Because of the volume, the correspondence is not attached. Copies of all comments received except those originated within the Commission have been placed in the Public Document Room.

4. The general reaction was that the criteria fulfilled a need and the AEC should continue their development. None of the correspondents objected to the issuance of general criteria and their comments were constructive. The Atomic Industrial Forum, for example, submitted a complete proposed revision reflecting considerable interest and effort on the part of that organization. The comments received fell into the following broad categories:

a. Title each criterion. This was suggested as an aid in indexing and referencing.

b. Improve the organization of the criteria. Comments included suggestions for arranging criteria according to type of systems and for grouping the criteria according to the degree of public protection.

c. Simplify the format. A number of suggestions were made for eliminating repetition for combining criteria and for clarification.

d. Eliminate details. Some comments suggested that the criteria should state only objectives, and that specific details and manner of implementation should not be stated. A number of comments expressed a desire for less general and for more comprehensive and detailed criteria.

e. Relate the criteria only to the protection of the public. Views were expressed that some criteria as written related to operational problems and should be eliminated.

f. Retitle the document. A belief was expressed that as written these were not truly criteria, but principles or fundamentals.

g. Apply the criteria more broadly than construction permits alone. This comment essentially urged that the restriction of the criteria to construction permits should be deleted and that they should be made applicable to all stages of licensing, including the operating license stage.

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[illegible] the criteria. In addition, subsequent redrafts were circulated to other divisions within the Commission. Principal comments from these divisions have been reflected in the revised criteria. Other comments from within the Commission will be considered in conjunction with public comments received after publication in the Federal Register.

6. The regulatory staff has worked closely with the Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment. The ACRS has stated that it believes that the revised criteria are appropriate to publish for public comment.

7. It is proposed that the criteria be included as Appendix A to 10 CFR 50. The proposed amendment, which is attached as Appendix "B," provides that the General Design Criteria be used for guidance by an applicant in developing the principal design criteria for the facility. For a specific reactor case, some of the General Design Criteria may be unnecessary or inappropriate and the criteria, as a whole, may be insufficient. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced reactor types. In any case, there must be assurance that the principal design criteria proposed by an applicant encompass all those facility design features required in the interest of public safety.

8. The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for Category B.

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9. The proposed General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

STAFF JUDGMENTS

10. The Office of the General Counsel and the Divisions of Reactor Licensing and Compliance concur in the recommendations of this paper. The Office of Congressional Relations concurs in Appendix "C." The Division of Public Information concurs in recommendation 11.c.

RECOMMENDATION

11. The Director of Regulation recommends that the Atomic Energy Commission:

a. Approve publication of the proposed amendments to 10 CFR Part 50 contained in Appendix "B."

b. Note that the Joint Committee on Atomic Energy will be informed by letter such as Appendix "C."

c. Note that a public announcement such as Appendix "D" be issued on filing the notice of proposed rule making with the Federal Register.

LIST OF ENCLOSURES

APPENDIX "A"

LIST OF INCOMING CORRESPONDENCE ON "AEC SEEKING PUBLIC COMMENT ON PROPOSED DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS" PRESS RELEASE NO. H-252 DATED NOVEMBER 22, 1965

1. J. B. McCarty, Jr., U.S. Coast Guard, 1/26/66.
2. E. P. Epler, Oak Ridge National Laboratory, 1/26/66.
3. Dr. Emerson Jones, Technical Management, Inc., 2/2/66.
4. H. C. Paxton and D. B. Hall, Los Alamos Scientific Laboratory, 2/2/66.
5. C. Starr, Atomics International, 2/4/66.
6. C. T. Chave, Stone and Webster Engineering Corporation, 2/11/66.
7. R. L. Junkins, Pacific Northwest Laboratory, 2/8/66.
8. Richard Hughes, Governor of New Jersey, 2/10/66.
9. Royce J. Rickert, Combustion Engineering, Inc., 2/11/66.
10. W. B. Cottrell, Oak Ridge National Laboratory, 2/11/66.
11. Peter A. Morris, Director, Division of Operational Safety, 2/11/66.
12. Holmes & Narver, Inc., 2/11/66.
13. CDR J. C. Ledoux, BuY&D, Dept. of Navy, 2/11/66.
14. Richard H. Peterson, Pacific Gas and Electric Company, 2/14/66.
15. Norbert L. Kopchinski, Professional Engineer, California, 2/14/66.
16. D. L. Crook, Dept. of Commerce, Maritime Adm., Wash., D.C., 2/15/66.
17. R. H. Harrison, Babcock & Wilcox, 2/22/66.
18. Theodore Stern, Westinghouse Electric Corporation, 2/25/66.
19. E. A. Wiggin, Atomic Industrial Forum, 2/28/66.
20. James G. Terrill, Jr., Dept. of Health, Education, and Welfare, Washington, D.C., 3/7/66.
21. J. P. Hogan, General Atomic, 4/30/66.
22. H. G. Rickover, Director, Division of Naval Reactors, 7/26/66.

APPENDIX "B"

[10 CFR PART 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria
for Nuclear Power Plant Construction Permits[1/]

The Atomic Energy Commission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits." The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

1/ Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) the programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from divisions within the Commission, from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

" (i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;"

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in §50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction

Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, United States Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

1. §50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§50.34 Contents of applications; technical information safety analysis report.[2]

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent

---

2/ Inasmuch as the Commission has under consideration other amendments to §50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of §50.34(b)(3)(i) previously published for comment in the FEDERAL REGISTER. /Additions are underscored./

subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

2. A new Appendix A is added to read as follows:

(See Attachment)

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at ______________________________ this ______________ day of ______________ 1967.

For the Atomic Energy Commission.

W. B. McCool
Secretary

GENERAL DESIGN CRITERIA FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS[3/]

Table of Contents

3/ Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

Group Title | Criterion No.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

V. REACTIVITY CONTROL

VI. REACTOR COOLANT PRESSURE BOUNDARY

VII. ENGINEERED SAFETY FEATURES

A. General Requirements for Engineered Safety Features

Group	Title	Criterion No.
VII.	ENGINEERED SAFETY FEATURES	
	B. Emergency Core Cooling Systems	
	Emergency Core Cooling Systems Capability	44
	Inspection of Emergency Core Cooling Systems	45
	Testing of Emergency Core Cooling Systems Components	46
	Testing of Emergency Core Cooling Systems	47
	Testing of Operational Sequence of Emergency Core Cooling Systems	48
	C. Containment	
	Containment Design Basis	49
	NDT Requirement for Containment Material	50
	Reactor Coolant Pressure Boundary Outside Containment	51
	Containment Heat Removal Systems	52
	Containment Isolation Valves	53
	Containment Leakage Rate Testing	54
	Containment Periodic Leakage Rate Testing	55
	Provisions for Testing of Penetrations	56
	Provisions for Testing of Isolation Valves	57
	D. Containment Pressure-Reducing Systems	
	Inspection of Containment Pressure-Reducing Systems	58
	Testing of Containment Pressure-Reducing Systems	59
	Testing of Containment Spray Systems	60
	Testing of Operational Sequence of Containment Pressure-Reducing Systems	61
	E. Air Cleanup Systems	
	Inspection of Air Cleanup Systems	62
	Testing of Air Cleanup Systems Components	63
	Testing of Air Cleanup Systems	64
	Testing of Operational Sequence of Air Cleanup Systems	65

Introduction

Every applicant for a construction permit is required by the provisions of §50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for Category B.

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable [illegible]

decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. NUCLEAR AND RADIATION CONTROLS

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B)

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. REACTIVITY CONTROL

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary [illegible] the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. ENGINEERED SAFETY FEATURES

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

CRITERION 46 - TESTING OF EMERGENCY CORE [illegible]

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30 F above nil ductility transition (NDT) temperature.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray

...as ... practical.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category B)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

APPENDIX "C"

[illegible]

1. Enclosed for the information of the Joint Committee on Atomic Energy is a Notice of Proposed Rule Making which would add to the proposed amendments to the Commission's regulations 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which were published in the Federal Register for comment on August 16, 1966. This amendment would add a new Appendix A to Part 50 "General Design Criteria for Nuclear Power Plant Construction Permits" to assist in the preparation of applications for construction permits for nuclear power plants.

2. The proposed change implements one of the key recommendations of the Regulatory Review Panel in which the Panel expressed the need for criteria to be used at the construction permit stage. As you know, work had been in progress on criteria development at the time of the Panel's recommendation. This effort was accelerated and led to the issuance of preliminary proposed criteria for public comment in Press Release H-252 dated November 22, 1965. The General Design Criteria included in the enclosed proposed amendment reflect comments and suggestions on the preliminary criteria received from industry, divisions within the Commission, the Advisory Committee on Reactor Safeguards, and the public.

3. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant as contemplated by the previously published revisions to Part 50. The framework within which the criteria are presented provides sufficient flexibility for applicants to establish design requirements using alternate and/or additional criteria so long as safety can be assured. In particular,

additional criteria will be needed for unusual sites and environmental conditions and for new or advanced types of reactors. In every case, however, the applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

4. The provisions of the proposed amendments relating to the General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

5. The notice of proposed rule making has been transmitted to the Office of the Federal Register for publication. Sixty days for public comment are provided. Enclosed also is a copy of an announcement we plan to issue in the next few days on this matter.

APPENDIX "D"

DRAFT PUBLIC ANNOUNCEMENT

AEC PUBLISHES GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The AEC is publishing for public comment a revised set of proposed General Design Criteria which have been developed to assist in the preparation of applications for nuclear power plant construction permits.

In November 1965, the AEC issued an announcement requesting comments on General Design Criteria developed by its regulatory staff. These criteria were statements of design principles and objectives which have evolved over the years in licensing nuclear power plants by the AEC.

It was recognized at the time the criteria were first issued for comment that further efforts were needed to develop them more fully. The revision being published today reflects comments received following the 1965 announcement, suggestions made at meetings with the Atomic Industrial Forum, and review within the AEC.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment.

The General Design Criteria reflect the predominating experience to date with water reactors, but they are considered to be generally applicable to all power reactors. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant. The framework within which the criteria are presented provides sufficient flexibility for applicants to establish design requirements using alternate and/or additional criteria so long as safety can be assured. In particular, additional criteria will be needed for unusual sites and environmental conditions and for new or advanced types of reactors. In every case, however, the applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for Category B.

Development of these criteria is part of a longer-range Commission program to develop criteria, standards, and codes for nuclear reactor plants. This includes codes and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry codes and standards based on accumulated knowledge and experience as has occurred in various fields of engineering and construction.

The provisions of the proposed amendment relating to General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

The proposed criteria, which would become Appendix A to Part 50 of the AEC's regulations, will be published in the Federal Register on ___________. Interested persons may submit written comments or suggestions to the Secretary, U. S. Atomic Energy Commission, Washington, D.C., 20545, within 60 days. A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

July 23, 1969

Dr. Stephen H. Hanauer, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Dr. Hanauer:

Enclosed are 18 copies of:

1. "General Design Criteria for Nuclear Power Units" revision dated July 15, 1969, which reflects the comments made by the ACRS Subcommittee at our meeting July 9, 1969, and

2. A "Comparison of Published Criteria (July 11, 1967) and Revised Criteria (July 15, 1969)."

Regarding the differences between the published and revised criteria, please note that the revised criteria:

a. Reflect comments received from industry on the published criteria and developments that have occurred since their release. In addition, they reflect comments received from the ACRS and the regulatory staff on interim drafts.

b. Establish "minimum requirements" for water-cooled reactors, whereas the published criteria were "guidance" for all reactors.

c. Are arranged in six sections, include definitions, and are not categorized (Category A or Category B).

d. Do not include the term "engineered safety features." The requirements in the published criteria for "engineered safety features" have been incorporated in the revised criteria by including the requirements in the criteria for individual systems.

e. Include criteria which do not have direct counterparts in the published criteria; these are located in the back of Enclosure 2.

ACRS review is requested as soon as possible.

Sincerely,

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
As stated

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

July 15, 1969

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER UNITS

Table of Contents

INTRODUCTION

Pursuant to the provisions of § 50.34, applications for construction permits must include the principal design criteria for a proposed facility. These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved for construction by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to be used for guidance in establishing the principal design criteria for these units.

The principal design criteria for a nuclear power unit establish necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that prevent or mitigate the consequences or accidents which could cause undue risk to the health and safety of the public. There will be some nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be established in the interest of public safety. It is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For units such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear reactor and associated equipment necessary for electrical power generation and those structures, systems, and components required to prevent or mitigate the consequences of accidents which could cause undue risk to the health and safety of the public.

REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary means all those pressure-containing components, such as pressure vessels, piping, pumps, and valves, within the following systems or portions of systems of pressurized and boiling water-cooled nuclear power units:

(a) The reactor coolant system. For a nuclear power unit of the boiling water type, the reactor coolant system extends to and includes the outermost containment isolation valves capable of external actuation in the main steam and feed-water lines, and the reactor safety and relief valves.

(b) Portions of associated auxiliary systems connected to the reactor coolant system. For piping of these systems which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside the containment capable of external actuation. For piping of these systems which contains two valves both of which are normally closed during normal reactor operation, the boundary extends to and includes the second of these two valves (the second of which must be capable of external actuation), whether or not the system piping penetrates primary reactor containment.

(c) Portions of the emergency core cooling system connected to the reactor coolant system. For piping of this system which penetrates primary reactor containment, the boundary extends to and includes the first containment isolation valve outside containment capable of external actuation. For piping of this system which does not penetrate primary reactor containment, the boundary extends to and includes the second of two valves normally closed during normal reactor operation.

LOSS-OF-COOLANT ACCIDENTS

Loss-of-coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from any size break in the piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and within the reactor coolant pressure boundary, up to and including a break in these components equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.

SINGLE FAILURE

A single failure means an occurrence which results in a loss of capability of a structure, system, or component to perform its intended functions. Multiple failures resulting from a single occurrence are considered to be a single failure.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Where generally recognized codes and standards are used, they shall be identified. Codes and standards shall be supplemented and modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power plant licensee throughout the life of the unit.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, (2) an appropriate margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (3) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (4) the importance of the safety function to be performed.

CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practicable throughout the unit particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the capability of these structures, systems, and components.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, testing, and postulated accidents. These structures, systems, and components shall be appropriately protected against dynamic effects and missiles that may result from equipment failures and sources outside the nuclear power unit.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel damage limits are not exceeded. The core and associated system designs shall assure this fuel integrity under all conditions of normal operation, including the effects of anticipated operational occurrences such as loss of power to recirculation pumps, coolant loss within the capability of the reactor coolant makeup system, tripping of a turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the effect of the inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can cause damage in excess of specified acceptable fuel damage limits are not possible or can be reliable and readily detected and suppressed.

CRITERION 13 - REACTOR INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to assure that variables and systems which can affect the fission process and the integrity of the reactor core are monitored and maintained within prescribed operating ranges.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary coolant, control, and protection systems shall be designed with appropriate margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded. The reactor coolant system and associated system designs shall assure these design conditions under all conditions of normal operation, including the effects of anticipated operational occurrences such as loss of power to the recirculation pumps, tripping of a turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment shall be provided. The containment and associated systems shall be designed to provide an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions are not exceeded for as long as any postulated accident condition requires.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

Onsite and offsite electrical power systems shall be provided with sufficient capacity and capability to assure that (1) specified acceptable fuel damage limits and design conditions of the reactor coolant pressure boundary are not exceeded during anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained following postulated accidents. The system safety function for the onsite and offsite electrical power systems shall be that each provide sufficient capacity to permit functioning of structures, systems, and components important to safety. Offsite electrical power shall be provided to the site preferably by two physically independent transmission lines. The onsite system and the onsite portions of the offsite system shall be designed with sufficient independency, redundancy, and testability to perform their safety function assuming failure of a single active component. Provisions shall be included to minimize the probability of losing offsite electrical power as a result of or coincident with the loss of electrical power generated by the nuclear power unit.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the system, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practicable, the full operational sequence that brings the system into operation, including the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection of the control room shall be provided to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accidents.

Equipment at appropriate locations outside the control room shall be provided (1) having a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable emergency procedures.

III. PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to act automatically to assure that specified acceptable fuel damage limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of redundancy. Means shall be included for testing the protection system when the reactor is in operation to determine failures and losses of redundancy and independence that may have occurred.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The effects of adverse conditions to which redundant channels of the protection system may be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as diversity in component design and principles of operation, shall be used to the extent practicable to prevent loss of the protection function in the event of systematic, non-random, concurrent failures of redundant elements.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL

The protection system shall be separated from control systems to the extent that failure or removal from service of any control system component or channel, or any one of those common to control and protection systems, leaves intact a system satisfying all reliability, redundancy, testability, and independence requirements for the protection system.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be capable of protecting against any single malfunction of the reactivity control systems, such as unplanned withdrawal (not ejection or dropout) of control rods or dilution of soluble poison, without exceeding acceptable fuel damage limits.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems, preferably of different design principles, shall be provided. Each system shall have the capability to control reactivity changes (including xenon burnout) resulting from planned, normal power changes without exceeding acceptable fuel damage limits. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel damage limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions the capability to cool the core is maintained.

CRITERION 28 - REACTIVITY LIMITS FOR ACCIDENTS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity insertion to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, changes in reactor coolant temperature and pressure, and cold water addition.

IV. FLUID SYSTEMS

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components within the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. Means shall be provided for detecting and, to the extent practicable, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The fracture toughness properties and the service temperatures of the reactor coolant pressure boundary shall assure nonbrittle behavior under operating, testing, and postulated accident conditions.

CRITERION 32 - DESIGN OF COMPONENTS WITHIN REACTOR COOLANT PRESSURE BOUNDARY

Components within the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing of important areas and features, including an appropriate material surveillance program for the reactor pressure vessel, to assess their structural and leaktight integrity.

CRITERION 33 - REACTOR COOLANT MAKEUP SYSTEM

A system to supply reactor coolant makeup during normal reactor operation, preferably through two system flow paths, shall be provided. The system safety function shall be to assure that specified acceptable fuel damage limits are not exceeded as a result of coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping within the boundary.

Redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless if can be demonstrated that the system is acceptable on some other defined basis.

CRITERION 34 - DECAY HEAT REMOVAL SYSTEM

A system to remove decay heat, preferably through two system flow paths, shall be provided. The system safety function shall be to transfer fission product decay heat and residual heat from the reactor core when the reactor is shutdown at a rate such that specified acceptable fuel damage limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless it can be demonstrated that the system is acceptable on some other defined basis.

CRITERION 35 - EMERGENCY CORE COOLING SYSTEM

A system to provide abundant emergency core cooling, preferably through two system flow paths and by different design principles, shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling are prevented and (2) clad metal-water reaction is limited to negligible amounts. The performance of the system shall be evaluated conservatively in each area of uncertainty.

Redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless it can be demonstrated that the system is acceptable on some other defined basis.

CRITERION 36 - DESIGN OF EMERGENCY CORE COOLING SYSTEM COMPONENTS

Components of the emergency core cooling system shall be designed to permit periodic inspection and testing of important areas and features, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the system, such as pumps and valves and (2) the operability of the system as a whole and, under conditions as close to design as practicable, the full operational sequence that brings the system into operation, including the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 38 - CONTAINMENT HEAT REMOVAL SYSTEM

A system to remove heat from the reactor containment, preferably through two system flow paths and by different design principles shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at low levels.

Redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless it can be demonstrated that the system is acceptable on some other defined basis.

CRITERION 39 - DESIGN OF CONTAINMENT HEAT REMOVAL SYSTEM COMPONENTS

Components of the containment heat removal system shall be designed to permit periodic inspection and testing of important areas and features, such as the torus, sumps, spray nozzles and piping, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the system, such as pumps and valves and (2) the operability of the system as a whole, and, under conditions as close to the design as practicable, the full operational sequence that brings the system into operation, including the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided. The systems safety functions shall be (1) to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following any postulated accident and (2) to control the concentration of hydrogen, oxygen, and other substances in the containment atmosphere following any postulated accident to assure that containment integrity is maintained.

Each system shall have redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities to assure that for onsite and for offsite electrical power system operation its safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless it can be demonstrated that the system is acceptable on some other defined basis.

CRITERION 42 - DESIGN OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS COMPONENTS

Components of the containment atmosphere cleanup systems shall be designed to permit periodic inspection of important areas and features such as filter frames, ducts, and piping to assure their structural and leaktight integrity and the full design capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (2) the operability of the systems as a whole and, under conditions as close to design as practicable, the full operational sequence that brings the systems into operation, including the transfer between normal and emergency power sources, and operation of associated systems.

CRITERION 44 - COOLING WATER SYSTEM

A system to transfer heat from structures, systems, and components important to safety, preferably through two system flow paths to the ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities shall be provided as required to assure that for onsite and for offsite electrical power system operation the system safety function can be accomplished assuming (1) failure of any single active component and (2) failure of any single passive component unless it can be demonstrated that the system is acceptable on some other basis.

CRITERION 45 - DESIGN OF COOLING WATER SYSTEM COMPONENTS

Components of the cooling water system shall be designed to permit periodic inspection of important areas and features, such as heat exchangers and piping, to assure their structural and leaktight integrity and the full design capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed with a capability to test periodically (1) the operability and functional performance of the active components of the system, such as pumps and valves, and (2) the operability of the system as a whole and, under conditions as close to design as practicable, and full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including the transfer between normal and emergency power sources.

V. REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and any necessary containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with an appropriate margin, the calculated peak pressure and temperature conditions resulting from any loss-of-coolant accident. This appropriate margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the potential that the effects of phenomena may be more severe than predicted, and (3) the limited experience and experimental data available for defining accident phenomena and containment response.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT

The fracture toughness properties and the service temperatures of the reactor containment ferritic materials shall assure nonbrittle behavior under operating, testing, and postulated accident conditions.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall have provisions (1) for inspection of all important areas, including penetrations, (2) for an appropriate materials surveillance program, and (3) for periodically testing the leaktightness of penetrations which have resilient seals and expansion bellows at containment design pressure.

CRITERION 54 - SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, testability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line which is part of the reactor coolant pressure boundary and which penetrates primary reactor containment shall be provided with one isolation valve inside and one isolation valve outside of containment. The valve outside of containment shall be located as close to containment as practicable. The primary mode for actuation of the valves shall be automatic and upon loss of actuating power these valves shall be designed to fail safe.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

CRITERION 56 - CONTAINMENT ATMOSPHERE ISOLATION VALVES

Each line which connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with two isolation valves. One of these valves shall be outside of containment and shall be located as close to containment as practicable. The primary mode for actuation of the valves shall be automatic and upon loss of actuating power these valves shall be designed to fail safe, unless it can be demonstrated that the system design is acceptable on some other defined basis.

CRITERION 57 - CLOSED SYSTEMS ISOLATION VALVES

Each line which penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one isolation valve. This valve shall be outside of containment and shall be located as close to containment as practicable.

VI. FUEL AND RADIOACTIVITY CONTROL

CRITERION 60 - FUEL STORAGE AND HANDLING AND RADIOACTIVE WASTE SYSTEMS

The fuel storage and handling and radioactive waste systems shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) to prevent significant reduction in fuel storage coolant inventory under accident conditions (2) with a decay heat removal capability having reliability, testability, and performance that reflect the importance to safety of decay heat removal, (3) with suitable shielding for radiation protection, (4) with a capability to permit inspection and testing of important areas and features of the components of these systems, and (5) with appropriate containment, confinement, and filtering systems.

CRITERION 61 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT

The nuclear power unit design shall include means to maintain suitable control over gaseous, liquid, and solid radioactive effluents that may be released from the unit during normal operations, anticipated operational occurrences, and postulated accidents. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, and solid effluents, particularly where unfavorable site environmental conditions can be expected to impose operational limitations upon the release of radioactive effluents.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by geometrically safe configurations.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Instrumentation shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of decay heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths and the unit environs for radioactivity that may be released from normal operations, from anticipated operational occurrences, and from postulated accidents.

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January 28, 1971

SECY-R 143

AMENDMENT TO 10 CFR 50 - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Note by the Secretary

The Director of Regulation has requested that the attached report by the Director of Reactor Standards be circulated for consideration by the Commission at an early Meeting.

W. B. McCool

Secretary of the Commission

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ATOMIC ENERGY COMMISSION
AMENDMENT TO 10 CFR 50
GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Report to the Director of Regulation
by the
Director, Division of Reactor Standards

THE PROBLEM

1. To consider publication in effective form of an amendment to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power Plants".

BACKGROUND AND SUMMARY

2. At Regulatory Meeting 255 on June 28, 1967, the Commission approved publication of a Notice of Proposed Rule Making to amend 10 CFR Part 50 by adding an Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" (AEC-R 2/57). That proposed amendment was published in the Federal Register on July 11, 1967, with a 60-day comment period.

3. Comments from twenty-one organizations and individuals, as listed in Appendix "B", were received in response to the previously proposed amendment. Because of the volume, the comments are not attached. Copies of all comments received have been placed in the Public Document Room.

4. The general reaction to the proposed criteria was favorable. The published proposed criteria were regarded as a considerable improvement over those originally released in Press Release H-252 dated November 22, 1965.* None of the commentators objected to the issuance of General Design Criteria. Most of the comments received were in the form of suggested improvements in language to facilitate understanding of the intent of the criteria, with few

*Secretariat Note: A copy of AEC Press Release H-252, November 22, 1965, is on file in the Office of the Secretary.

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suggestions to change or delete many requirements. The more significant comments and our resolution of them were:

a. Published Criterion 1 - Quality Standards

Comment - It should not be necessary for each applicant to show that an applicable code or standard is sufficient. A showing of sufficiency should be required only for those items not covered by an applicable code or standard.

Resolution - This criterion has been modified to provide that a showing of sufficiency is not necessarily required, but an evaluation by the applicant of the applicable codes and standards to determine sufficiency is necessary (see New Criterion 1). Nuclear codes and standards have not been developed to the degree where it can be assumed that they are sufficient. The number of codes that has remained in an "Issued for Trial Use and Comment" status for long periods of time and the additional requirements contained in the addenda to accepted codes indicate the need for an applicant to evaluate applicable codes and standards to assure their sufficiency.

b. Published Criterion 11 - Control Room

Comments - (1) The criterion as published could be interprated to require two control rooms and (2) Part 20 is not applicable to accidents.

Resolution - The criterion has been rewritten to make it clear that only one control room is required and reference to Part 20 has been deleted (see New Criterion 19). It should be noted that we have discussed control

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room requirements with industry representatives in order to understand better their views. One reactor manufacturer, supported by several utilities, made a presentation to the regulatory staff on this subject. The new wording of the criterion is in agreement with the industry position expressed in these discussions.

c. Published Criterion 28 - Reactivity Hot Shutdown Capability

Comment - The criterion can be interpreted to require two reactivity control systems capable of fast shutdown.

Resolution - The criterion has been rewritten to make it clear that only one system must be capable of fast shutdown (see New Criterion 26).

d. Published Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention

Comment - The requirements of this criterion are too specific and should be deleted.

Resolution - The criterion has been rewritten in a more general form. All references to specific margins above NDT temperature have been deleted (see New Criterion 31). Interim draft revisions of the criterion on fracture prevention were discussed with the major reactor manufacturers. This resulted in a change in their position from recommending that the criterion be deleted to recommending that it be retained in the revised form.

e. Published Criterion 39 - Emergency Power for Engineered Safety Features

Comment - (1) The requirement that offsite power must satisfy the "single failure criterion" is impractical and (2) eliminate all reference to offsite power.

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Resolution - The criterion has been rewritten to make it clear that the offsite power system need not meet the "single failure criterion." Reference to offsite power has not been deleted because we believe that offsite power is required to provide adequate assurance of safety (see New Criterion 17). New Criterion 17 has been discussed with the IEEE Subcommittee which is developing criteria for power requirements for nuclear power units. The members of the subcommittee indicated that the new criterion is acceptable and consistent with their requirements.

f. Published Criterion 44 - Emergency Core Cooling Systems Capability

Comment - Two independent emergency core cooling systems are not necessary.

Resolution - The criterion has been rewritten so that one system with sufficient redundancy is acceptable (see New Criterion 35). An interim version of the revised criterion for emergency core cooling was discussed with the ANS Systems Engineering Subcommittee. This subcommittee is in the process of developing criteria applicable to pressurized-water reactors. This interim version, which presented the one system concept, was acceptable to the ANS group with minor suggestions for changes in wording.

g. Published Criterion 49 - Containment Design Basis

Comment - Functioning of the emergency core cooling system is required for containment integrity; therefore,

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it is inconsistent to require that the containment design be based on the assumed failure of emergency core cooling systems.

Resolution - The criterion has been rewritten so that for containments a design margin which reflects consideration of the possible effects of degraded emergency core cooling performance is required (see New Criterion 50).

5. The staff met in February 1970 with an ad hoc AIF group, which included representatives of reactor manufacturers, utilities and architect engineers to discuss the revised General Design Criteria. The comments of this group were reflected in a June 4, 1970 draft of the revised General Design Criteria that was forwarded to the AIF for comment. The AIF forwarded comments and stated it believed the criteria should be published as an effective rule after reflecting its comments. These comments have been reflected in the General Design Criteria in Appendix "A".

6. The revised criteria establish minimum requirements for the design of water-cooled nuclear power units and provide guidance for the design of other nuclear power units whereas the previously proposed criteria provided guidance for applicants for construction permits for all types of nuclear power plants.

7. The revised criteria include definitions in accordance with comments received from industry that certain crucial terms should be defined. In addition, the criteria have been rearranged to increase their usefulness to designers and evaluators.

8. The Category A or B designation for each criterion which was included in the previously proposed amendment has been deleted. These categories had been included to provide guidance on the quantity and detail of information required for individual items at the construction permit stage. The amendment to § 50.34 of 10 CFR Part 50, published December 17, 1968, gives sufficient guidance in this area.

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9. The revised criteria do not include the term "engineered safety features." The requirements in the previously proposed criteria for these features have been incorporated in the revised criteria for the individual systems which are used for this purpose.

10. There are new criteria which do not have direct counterparts in the previously proposed criteria. Most of these do not represent new requirements but represent more specific guidance on requirements that were included in the previously proposed criteria in a more general form.

11. The regulatory staff has considered all comments received in revising the criteria and has worked closely with the Advisory Committee on Reactor Safeguards in the development of the criteria. The criteria in Appendix "A" reflect ACRS review and comments.

STAFF JUDGMENTS

12. The Divisions of Reactor Licensing and Compliance and the Office of the General Counsel concur in the recommendation of this paper. The draft public announcement was prepared by the Division of Public Information. The Office of Congressional Relations concurs in the draft letter to the Joint Committee on Atomic Energy.

RECOMMENDATION

13. The Director of Regulation recommends that the Atomic Energy Commission:

a. Approve publication in effective form of the amendment to 10 CFR Part 50 which would add an Appendix A, "General Design Criteria for Nuclear Power Plants" establishing minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been previously issued by the Commission and providing guidance to the applicants for construction permits for establishing the principal design criteria for other types of nuclear power plants;

b. Note that the amendment to 10 CFR Part 50 set forth in Appendix "A" will be published in the Federal Register to be effective 90 days after publication.

c. Note that the Joint Committee on Atomic Energy will be informed by letter such as Appendix "C";

d. Note that a public announcement such as Appendix "D" will be issued when the amendment is filed with the Federal Register.

LIST OF ENCLOSURES

APPENDIX "A"

TITLE 10 - ATOMIC ENERGY

CHAPTER 1 - ATOMIC ENERGY COMMISSION

PART 50 - LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

The Atomic Energy Commission has adopted an amendment to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which adds an Appendix A, "General Design Criteria for Nuclear Power Plants."

Paragraph 50.34(a) of Part 50 requires that each application for a construction permit include the preliminary design of the facility. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility

(ii) The design bases and the relation of the design bases to the principal design criteria

(iii) Information relative to materials of construction, general arrangement, and the approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" added as Appendix A to Part 50 establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified.

A proposed Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" to 10 CFR Part 50 was published in the FEDERAL REGISTER (32 FR 10213) on July 11, 1967. The comments and suggestions received in response to the notice of proposed rule making and subsequent developments in the technology and in the licensing process have been considered in developing the revised criteria which follow.

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, non-random, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

[illegible] of [illegible], as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U. S. Atomic Energy Commission, Washington, D. C., 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

1. Subdivision 50.34(a)(3)(i) is amended to read as follows:

§ 50.34 Contents of applications; technical information.

(a) Preliminary safety analysis report. Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility. Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to [illegible] construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units;

* * * * *

2. Footnote[2] to § 50.34 is amended to read as follows:

[2]General design criteria for chemical processing facilities are being developed.

* * * * *

3. A new Appendix A is added to read as follows:

APPENDIX A

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

[illegible] to [illegible] failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26 and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

NUCLEAR POWER UNIT

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

LOSS OF COOLANT ACCIDENTS

Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

SINGLE FAILURE

A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

ANTICIPATED OPERATIONAL OCCURRENCES

Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited

[1]Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2]Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as [illegible], hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 10 - REACTOR DESIGN

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

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power sources, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electrical power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electrical power sources.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRICAL POWER SYSTEMS

Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment [illegible] outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. PROTECTION AND REACTIVITY CONTROL SYSTEMS

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable [illegible] of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems as the simultaneous failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, [illegible], and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. FLUID SYSTEMS

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

[illegible] 32 - INSP[illegible]

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. REACTOR CONTAINMENT

CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating [illegible] The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment. or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment. or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment. or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. FUEL AND RADIOACTIVITY CONTROL

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Dated at ______________________________ this __________

day of ________________ 1971.

FOR THE ATOMIC ENERGY COMMISSION

____
W. B. McCool
Secretary of the Commission

APPENDIX "B"

LIST OF COMMENTS ON
PREVIOUS NOTICE OF PROPOSED RULE MAKING (32 FR 10213)
PUBLISHED IN THE FEDERAL REGISTER, JULY 11, 1967

1. H. C. Paxton, Los Alamos Scientific Laboratory, Member ASLB Panel 7/25/67.
2. Eugene Greuling, Duke University Member, ASLB Panel, 7/26/67.
3. Stuart McLain, McLain Associates, 8/22/67.
4. Einar Svanson, Black and Veatch, 8/25/67.
5. G. J. Stathakis, General Electric Company, 9/5/67.
6. William B. Cottrell, Oak Ridge National Laboratory, 9/6/67.
7. J. M. Gallagher, Jr., IEEE, Nuclear Science Group, Reactor Instrumentation and Controls Standards Subcommittee, 9/6/67.
8. David N. Barry, III, Southern California Edison Company, 9/7/67.
9. J. C. Rengel, Westinghouse Electric Corporation, 9/8/67.
10. W. B. Behnke Jr., Commonwealth Edison Company, 9/8/67.
11. Sol Burstein, Wisconsin Electric Power Company, 9/8/67.
12. L. E. Minnick, Yankee Atomic Electric Company, 9/8/67.
13. D. M. Leppke, Pioneer Service and Engineering Company, 9/19/67.
14. W. R. Cooper, Tennessee Valley Authority, 9/20/67.
15. R. E. Wascher, Babcock & Wilcox, 9/20/67.
16. J. J. Flaherty, Atomics International, 9/25/67.
17. Edwin A. Wiggin, Atomics Industrial Forum, Inc., 10/2/67.
18. William S. Lee, Duke Power Company 11/2/67.
19. Charles O'D. Lee, Jr., Specifications Engineer, California, 12/20/6
20. H. B. Stewart, Gulf General Atomic, Inc., 2/15/68.
21. J. M. West, Combustion Engineering, Inc., 2/21/68.

APPENDIX "C"

DRAFT LETTER TO THE JOINT COMMITTEE ON ATOMIC ENERGY

1. Enclosed for the information of the Joint Committee is a copy of a notice of rule making amending the Commission's regulation "Licensing of Production and Utilization Facilities," 10 CFR Part 50 to add an Appendix A, General Design Criteria for Nuclear Power Plants. Proposed criteria were published for comment on July 11, 1967. The criteria in the notice of rule making reflect consideration of the comments received on the proposed criteria published for comment and subsequent developments in the technology and in the licensing process.

2. The criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission. They also provide guidance to applicants for construction permits for establishing the principal design criteria for other types of nuclear power plants.

3. The amendment will be effective 90 days after publication in the Federal Register.

4. Enclosed also is a copy of a public announcement we plan to issue on this matter in the next few days.

APPENDIX "D"

DRAFT PUBLIC ANNOUNCEMENT

AEC PUBLISHES GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

The AEC is publishing a revised set of general design criteria for use in establishing the principal design criteria for nuclear power plants.

In July 1967 AEC published in the Federal Register for public comment "General Design Criteria for Nuclear Power Plant Construction Permits" developed by its regulatory staff. The revision published today reflects extensive comment received from 21 groups or individuals, review within the AEC, and developments that have occurred in the nuclear industry since publication of the criteria in 1967.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards in developing the revised criteria.

The amendment to Part 50 of the Commission's regulations fixes minimum requirements for the principal design criteria for water-cooled nuclear power units similar in design and location to units previously approved by the Commission for construction. It provides guidance, also, for establishing the principal design criteria for other

Federal Register will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Birmingham 1,200-foot transition area described in § 71.181 (32 F.R. 2148 and 3765) would be altered as follows:

"* * * thence southwest along the southeast boundary of V-209 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VOR; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincide with R-2101 and the Gadsden, Ala., transition area * * *." would be deleted and "* * * thence southwest along the southeast boundary of V-209 to longitude 88°00'00" W.; thence north along longitude 88°00'00" W. to the north boundary of V-18; thence northeast along the north boundary of V-18 to a 19-mile radius arc centered on the Tuscaloosa, Ala., VORTAC; thence clockwise along this arc to longitude 87°30'00" W.; thence north along longitude 87°30'00" W. to point of beginning, excluding that portion that coincides with R-2101 and the Gadsden, Ala., transition area * * *." would be substituted therefor.

The proposed additional airspace is required for the protection of IFR operations and for radar vectoring of aircraft arriving and departing the Birmingham area.

The official docket will be available for examination by interested persons at the Southern Regional Office, Federal Aviation Administration, Room 724, 3400 Whipple Street, East Point, Ga.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 30, 1967.

JAMES G. ROGERS,
Director, Southern Region.

[F.R. Doc. 67-7949; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67-SO-64]

TRANSITION AREA

Proposed Designation

The Federal Aviation Administration is considering an amendment to Part 71 of the Federal Aviation Regulations that would designate the Camden, S.C., transition area.

Interested persons may submit such written data, views, or arguments as they may desire. Communications should be submitted in triplicate to the Area Manager, Atlanta Area Office, Attention: Chief, Air Traffic Branch, Federal Aviation Administration, Post Office Box 20636, Atlanta, Ga. 30320. All communications received within 30 days after publication of this notice in the Federal Register will be considered before action is taken on the proposed amendment. No hearing is contemplated at this time, but arrangements for informal conferences with Federal Aviation Administration officials may be made by contacting the Chief, Air Traffic Branch. Any data, views, or arguments presented during such conferences must also be submitted in writing in accordance with this notice in order to become part of the record for consideration. The proposal contained in this notice may be changed in the light of comments received.

The Camden transition area would be designated as:

> That airspace extending upward from 700 feet above the surface within a 7-mile radius of Woodward Field (latitude 34°17'03" N., longitude 80°33'53" W.); within 2 miles each side of the 040° bearing from the Camden RBN (latitude 34°17'02" N., longitude 80°33'42.5" W.), extending from the 7-mile radius area to 8 miles northeast of the RBN.

The proposed transition area is required for the protection of IFR operations at Woodward Field. A prescribed instrument approach procedure to this airport utilizing the Camden (private) nondirectional radio beacon is proposed in conjunction with the designation of this transition area.

This amendment is proposed under section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348(a)).

Issued in East Point, Ga., on June 21, 1967.

GORDON A. WILLIAMS, Jr.
Acting Director, Southern Region.

[F.R. Doc. 67-7950; Filed, July 10, 1967; 8:49 a.m.]

[14 CFR Part 71]

[Airspace Docket No. 67-EA-1]

FEDERAL AIRWAYS

Supplemental Proposed Alteration

On March 1, 1967, a notice of proposed rule making was published in the Federal Register (32 F.R. 3402) stating that the Federal Aviation Agency was considering amendments to Part 71 of the Federal Aviation Regulations that would realign V-1 from Cape Charles, Va., via the INT of Cape Charles 013° and Salisbury, Md., 206° True radials; to Salisbury; that would designate a segment of V-139 from Norfolk, Va., via Cape Charles; to Snow Hill, Md., including a west alternate from Norfolk to Snow Hill via INT of Norfolk 360° and Snow Hill 226° True radials; and that would revoke the segment of V-194 from Norfolk to INT of Norfolk 001° and Cape Charles 313° True radials. Floors of 1,200 feet above the surface were proposed for these airway segments. These actions were proposed to simplify air traffic control procedures and flight planning in the Norfolk area.

Subsequent to publication of the notice, it was determined that the Snow Hill 226° True radial would not support a Federal airway. Accordingly, the proposals published in the notice are hereby cancelled and in lieu thereof, consideration is given to the following airway alignments that would serve the same purpose.

1. Redesignate the segment of V-194 from Norfolk via the intersection of Norfolk 001° T (008° Mag.) and Harcum, Va., 072° T (079° Mag.) radials; to the intersection of Harcum 072° and Snow Hill 211° True radials.

2. Realign V-1 from Cape Charles via the intersection of Cape Charles 009° T (016° Mag.) and Salisbury 206° T (214° Mag.) radials; to Salisbury.

Interested persons may participate in the proposed rule making by submitting such written data, views, or arguments as they may desire. Communications should identify the airspace docket number and be submitted in triplicate to the Director, Eastern Region, Attention: Chief, Air Traffic Division, Federal Aviation Administration, Federal Building, John F. Kennedy International Airport, Jamaica, N.Y. 11430. All communications received within 45 days after publication of this notice in the Federal Register will be considered before action is taken on the proposed amendment. The proposal contained in this notice may be changed in the light of comments received.

An official docket will be available for examination by interested persons at the Federal Aviation Administration, Office of the General Counsel, Attention: Rules Docket, 800 Independence Avenue SW., Washington, D.C. 20590. An informal docket will be available for examination at the office of the Regional Air Traffic Division Chief.

These amendments are proposed under the authority of section 307(a) of the Federal Aviation Act of 1958 (49 U.S.C. 1348).

Issued in Washington, D.C., on July 3, 1967.

T. McCORMACK,
Acting Chief, Airspace and Air Traffic Rules Division.

[F.R. Doc. 67-7951; Filed, July 10, 1967; 8:49 a.m.]

ATOMIC ENERGY COMMISSION

[10 CFR Part 50]

LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plant Construction Permits

The Atomic Energy Comission has under consideration an amendment to its regulation, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would add an Appendix A, "General Design Criteria for Nuclear Power

Plant Construction Permits."[1] The purpose of the proposed amendment would be to provide guidance to applicants in developing the principal design criteria to be included in applications for Commission construction permits. These General Design Criteria would not add any new requirements, but are intended to describe more clearly present Commission requirements to assist applicants in preparing applications.

The proposed amendment would complement other proposed amendments to Part 50 which were published for public comment in the FEDERAL REGISTER on August 16, 1966 (31 F.R. 10891).

The proposed amendments to Part 50 reflect a recommendation made by a seven-member Regulatory Review Panel, appointed by the Commission to study: (1) The programs and procedures for the licensing and regulation of reactors and (2) the decision-making process in the Commission's regulatory program. The Panel's report recommended the development, particularly at the construction permit stage of a licensing proceeding, of design criteria for nuclear power plants. Work on the development of such criteria had been in process at the time of the Panel's study.

As a result, preliminary proposed criteria for the design of nuclear power plants were discussed with the Commission's Advisory Committee on Reactor Safeguards and were informally distributed for public comment in Commission Press Release H-252 dated November 22, 1965. In developing the proposed criteria set forth in the proposed amendments to Part 50, the Commission has taken into consideration comments and suggestions from the Advisory Committee on Reactor Safeguards, from members of industry, and from the public.

Section 50.34, paragraph (b), as published for comment in the FEDERAL REGISTER on August 16, 1966, would require that each application for a construction permit include a preliminary safety analysis report. The minimum information to be included in this preliminary safety analysis report is (1) a description and safety assessment of the site, (2) a summary description of the facility, (3) a preliminary design of the facility, (4) a preliminary safety analysis and evaluation of the facility, (5) an identification of subjects expected to be technical specifications, and (6) a preliminary plan for the organization, training, and operation. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility;

(ii) The design bases and the relation of the design bases to the principal design criteria;

(iii) Information relative to materials of construction, general arrangement and approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety;

The "General Design Criteria for Nuclear Power Plant Construction Permits" proposed to be included as Appendix A to this part are intended to aid the applicant in development item (i) above, the principal design criteria. All criteria established by an applicant and accepted by the Commission would be incorporated by reference in the construction permit. In considering the issuance of an operating license under the regulations, the Commission would assure that the criteria had been met in the detailed design and construction of the facility or that changes in such criteria have been justified.

Section 50.34 as published in the FEDERAL REGISTER on August 16, 1966, would be further amended by adding to Part 50 a new Appendix A containing the General Design Criteria applicable to the construction of nuclear power plants and by a specific reference to this Appendix in § 50.34, paragraph (b).

The Commission expects that the provisions of the proposed amendments relating to General Design Criteria for Nuclear Power Plant Construction Permits will be useful as interim guidance until such time as the Commission takes further action on them.

Pursuant to the Atomic Energy Act of 1954, as amended, and the Administrative Procedure Act of 1946, as amended, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated. All interested persons who desire to submit written comments or suggestions in connection with the proposed amendments should send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, within 60 days after publication of this notice in the FEDERAL REGISTER. Comments received after that period will be considered if it is practicable to do so, but assurance of consideration cannot be given except as to comments filed within the period specified. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

1. Section 50.34(b)(3)(i) of 10 CFR Part 50 is amended to read as follows:

§ 50.34 Contents of applications; technical information safety analysis report.[2]

* * * * *

(b) Each application for a construction permit shall include a preliminary safety analysis report. The report shall cover all pertinent subjects specified in paragraph (a) of this section as fully as available information permits. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility, including:

(i) The principal design criteria for the facility. Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.

* * * * *

2. A new Appendix A is added to read as follows:

[2] Inasmuch as the Commission has under consideration other amendments to § 50.34 (31 F.R. 10891), the amendment proposed herein would be a further revision of § 50.34 (b)(3)(i) previously published for comment in the FEDERAL REGISTER.

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS[1]

TABLE OF CONTENTS

[1] Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

Group and title	Criterion No.
V. Reactivity control:	
Redundancy of Reactivity Control	27
Reactivity Hot Shutdown Capability	28
Reactivity Shutdown Capability	29
Reactivity Holddown Capability	30
Reactivity Control Systems Malfunction	31
Maximum Reactivity Worth of Control Rods	32
VI. Reactor coolant pressure boundary:	
Reactor Coolant Pressure Boundary Capability	33
Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention	34
Reactor Coolant Pressure Boundary Brittle Fracture Prevention	35
Reactor Coolant Pressure Boundary Surveillance	36
VII. Engineered safety features:	
A. General requirements for engineered safety features:	
Engineered Safety Features Basis for Design	37
Reliability and Testability of Engineered Safety Features	38
Emergency Power for Engineered Safety Features	39
Missile Protection	40
Engineered Safety Features Performance Capability	41
Engineered Safety Features Components Capability	42
Accident Aggravation Prevention	43
B. Emergency core cooling systems:	
Emergency Core Cooling Systems Capability	44
Inspection of Emergency Core Cooling Systems	45
Testing of Emergency Core Cooling Systems Components	46
Testing of Emergency Core Cooling Systems	47
Testing of Operational Sequence of Emergency Core Cooling Systems	48
C. Containment:	
Containment Design Basis	49
NDT Requirement for Containment Material	50
Reactor Coolant Pressure Boundary Outside Containment	51
Containment Heat Removal Systems	52
Containment Isolation Valves	53
Containment Leakage Rate Testing	54
Containment Periodic Leakage Rate Testing	55
Provisions for Testing of Penetrations	56
Provisions for Testing of Isolation Valves	57
D. Containment pressure-reducing systems:	
Inspection of Containment Pressure-Reducing Systems	58
Testing of Containment Pressure-Reducing Systems	59
Testing of Containment Spray Systems	60
Testing of Operational Sequence of Containment Pressure-Reducing Systems	61
E. Air cleanup systems:	
Inspection of Air Cleanup Systems	62
Testing of Air Cleanup Systems Components	63
Testing of Air Cleanup Systems	64
Testing of Operational Sequence of Air Cleanup Systems	65
VIII. Fuel and waste storage systems:	
Prevention of Fuel Storage Criticality	66
Fuel and Waste Storage Decay Heat	67
Fuel and Waste Storage Radiation Shielding	68
Protection Against Radioactivity Release from Spent Fuel and Waste Storage	69
IX. Plant effluents:	
Control of Releases of Radioactivity to the Environment	70

[1] Inasmuch as the Commission has under consideration other amendments to 10 CFR Part 50 (31 F.R. 10891), the amendment proposed herein would be a further revision to Part 50 previously published for comment in the FEDERAL REGISTER.

Introduction. Every applicant for a construction permit is required by the provisions of § 50.34 to include the principal design criteria for the proposed facility in the application. These General Design Criteria are intended to be used as guidance in establishing the principal design criteria for a nuclear power plant. The General Design Criteria reflect the predominating experience with water power reactors as designed and located to date, but their applicability is not limited to these reactors. They are considered generally applicable to all power reactors.

Under the Commission's regulations, an applicant must provide assurance that its principal design criteria encompass all those facility design features required in the interest of public health and safety. There may be some power reactor cases for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. There will be other cases in which these criteria are insufficient, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional criteria will be needed particularly for unusual sites and environmental conditions, and for new and advanced types of reactors. Within this context, the General Design Criteria should be used as a reference allowing additions or deletions as an individual case may warrant. Departures from the General Design Criteria should be justified.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information is needed at the construction permit stage for the items listed in Category A than for those in Category B.

I. OVERALL PLANT REQUIREMENTS

Criterion 1—Quality Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Criterion 2—Performance Standards (Category A). Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) Appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Criterion 3—Fire Protection (Category A). The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Criterion 4—Sharing of Systems (Category A). Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Criterion 5—Records Requirements (Category A). Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

Criterion 6—Reactor Core Design (Category A). The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

Criterion 7—Supression of Power Oscillations (Category B). The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Criterion 8—Overall Power Coefficient (Category B). The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Criterion 9—Reactor Coolant Pressure Boundary (Category A). The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Criterion 10—Containment (Category A). Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. NUCLEAR AND RADIATION CONTROLS

Criterion 11—Control Room (Category B). The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Criterion 12—Instrumentation and Control Systems (Category B). Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Criterion 13—Fission Process Monitors and Controls (Category B). Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Criterion 14—Core Protection Systems (Category B). Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Criterion 15—Engineered Safety Features Protection Systems (Category B). Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Criterion 16—Monitoring Reactor Coolant Pressure Boundary (Category B). Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Criterion 17—Monitoring Radioactivity Releases (Category B). Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

Criterion 18—Monitoring Fuel and Waste Storage (Category B). Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

Criterion 19—Protection Systems Reliability (Catergory B). Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

Criterion 20—Protection Systems Redundancy and Independence (Category B). Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Criterion 21—Single Failure Definition (Category B). Multiple failures resulting from a single event shall be treated as a single failure.

Criterion 22—Separation of Protection and Control Instrumentation Systems (Category B). Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Criterion 23—Protection Against Multiple Disability for Protection Systems (Category B). The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

Criterion 24—Emergency Power for Protection Systems (Category B). In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

Criterion 25—Demonstration of Functional Operability of Protection Systems (Category B). Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Criterion 26—Protection Systems Fail-Safe Design (Category B). The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. REACTIVITY CONTROL

Criterion 27—Redundancy of Reactivity Control (Category A). At least two independent reactivity control systems, preferably of different principles, shall be provided.

Criterion 28—Reactivity Hot Shutdown Capability (Category A). At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Criterion 29—Reactivity Shutdown Capability (Category A). At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Criterion 30—Reactivity Holddown Capability (Category B). At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Criterion 31—Reactivity Control Systems Malfunction (Category B). The reactivity control systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Criterion 32—Maximum Reactivity Worth of Control Rods (Category A). Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. REACTOR COOLANT PRESSURE BOUNDARY

Criterion 33—Reactor Coolant Pressure Boundary Capability (Category A). The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Criterion 34—Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A). The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Criterion 35—Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A). Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120° F. above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60° F. above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Criterion 36—Reactor Coolant Pressure Boundary Surveillance (Category A). Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. ENGINEERED SAFETY FEATURES

Criterion 37—Engineered Safety Features Basis for Design (Category A). Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Criterion 38—Reliability and Testability of Engineered Safety Features (Category A). All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Criterion 39—Emergency Power for Engineered Safety Features (Category A). Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Criterion 40—Missile Protection (Category A). Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Criterion 41—Engineered Safety Features Performance Capability (Category A). Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Criterion 42—Engineered Safety Features Components Capability (Category A). Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Criterion 43—Accident Aggravation Prevention (Category A). Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Criterion 44—Emergency Core Cooling Systems Capability (Category A). At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Criterion 45—Inspection of Emergency Core Cooling Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

Criterion 46—Testing of Emergency Core Cooling Systems Components (Category A). Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 47—Testing of Emergency Core Cooling Systems (Category A). A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

Criterion 48—Testing of Operational Sequence of Emergency Core Cooling Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

Criterion 49—Containment Design Basis (Category A). The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

Criterion 50—NDT Requirement for Containment Material (Category A). Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30° F. above nil ductility transition (NDT) temperature.

Criterion 51—Reactor Coolant Pressure Boundary Outside Containment (Category A). If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

Criterion 52—Containment Heat Removal Systems (Category A). Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Criterion 53—Containment Isolation Valves (Category A). Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

Criterion 54—Containment Leakage Rate Testing (Category A). Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

Criterion 55—Containment Periodic Leakage Rate Testing (Category A). The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Criterion 56—Provisions for Testing of Penetrations (Category A). Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Criterion 57—Provisions for Testing of Isolation Valves (Category A). Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

Criterion 58—Inspection of Containment Pressure-Reducing Systems (Category A). Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

Criterion 59—Testing of Containment Pressure-Reducing Systems Components (Category A). The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

Criterion 60—Testing of Containment Spray Systems (Category A). A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

Criterion 61—Testing of Operational Sequence of Containment Pressure-Reducing Systems (Category A). A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

Criterion 62—Inspection of Air Cleanup Systems (Category A). Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

Criterion 63—Testing of Air Cleanup Systems Components (Category A). Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Criterion 64—Testing of Air Cleanup Systems (Category A). A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

Criterion 65—Testing of Operational Sequence of Air Cleanup Systems (Category A). A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. Fuel and Waste Storage Systems

Criterion 66—Prevention of Fuel Storage Criticality (Category B). Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Criterion 67—Fuel and Waste Storage Decay Heat (Category B). Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Criterion 68—Fuel and Waste Storage Radiation Shielding (Category B). Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

Criterion 69—Protection Against Radioactivity Release From Spent Fuel and Waste Storage (Category B). Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

Criterion 70—Control of Releases of Radioactivity to the Environment (Category B). The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D.C., this 28th day of June 1967.

For the Atomic Energy Commission.

W. B. McCool,
Secretary.

[F.R. Doc. 67-7901; Filed, July 10, 1967; 8:45 a.m.]

Act of February 2, 1903, as amended the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, the reference to the State of Ohio in the introductory portion of paragraph (e) and paragraph (e)(9) relating to the State of Ohio are deleted.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended.)

Effective date. The foregoing amendment shall become effective upon issuance.

The amendment excludes a portion of Clinton County, Ohio, from the areas quarantined because of hog cholera. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not apply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the excluded area. No areas in Ohio remain under the quarantine.

The amendment relieves certain restrictions presently imposed but no longer deemed necessary to prevent the spread of hog cholera and must be made effective immediately to be of maximum benefit to affected persons. It does not appear that public participation in this rule making proceeding would make additional information available to this Department. Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendment are impracticable and unnecessary, and good cause is found for making it effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2380 Filed 2-19-71;8:49 am]

[Docket No. 71-526]

PART 76—HOG CHOLERA AND OTHER COMMUNICABLE SWINE DISEASES

Areas Quarantined

Pursuant to provisions of the Act of May 29, 1884, as amended, the Act of February 2, 1903, as amended, the Act of March 3, 1905, as amended, the Act of September 6, 1961, and the Act of July 2, 1962 (21 U.S.C. 111-113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f), Part 76, Title 9, Code of Federal Regulations, restricting the interstate movement of swine and certain products because of hog cholera and other communicable swine diseases, is hereby amended in the following respects:

In § 76.2, in paragraph (e)(13) relating to the State of Texas, subdivision (xvi) relating to Smith County is deleted, and new subdivisions (xxii) and (xxiii) relating to Bexar County are added to read:

(13) *Texas.* * * *

(xxii) That portion of Bexar County bounded by a line beginning at the junction of Interstate Highway 410 and Farm-to-Market Road 78; thence, following Farm-to-Market Road 78 in a northeasterly direction to Farm-to-Market Road 1518; thence, following Farm-to-Market Road 1518 in a southeasterly and then southwesterly direction to U.S. Highway 87; thence, following U.S. Highway 87 in a northwesterly direction to Interstate Highway 410; thence, following Interstate Highway 410 in a northwesterly direction to its junction with Farm-to-Market Road 78.

(xxiii) That portion of Bexar County bounded by a line beginning at the junction of the Bexar-Medina County line and State Highway 16; thence, following State Highway 16 in a southeasterly direction to Farm-to-Market Road 471; thence, following Farm-to-Market Road 471 in a southwesterly and then northwesterly direction to Farm-to-Market Road 1957; thence, following Farm-to-Market Road 1957 in a southeasterly and then southwesterly direction to the Bexar-Medina County line; thence, following the Bexar-Medina County line in a northerly direction to its junction with State Highway 16.

(Secs. 4-7, 23 Stat. 32, as amended, secs. 1, 2, 32 Stat. 791-792, as amended, secs. 1-4, 33 Stat. 1264, 1265, as amended, sec. 1, 75 Stat. 481, secs. 3 and 11, 76 Stat. 130, 132; 21 U.S.C. 111, 112, 113, 114g, 115, 117, 120, 121, 123-126, 134b, 134f; 29 F.R. 16210, as amended)

Effective date. The foregoing amendments shall become effective upon issuance.

The amendments quarantine portions of Bexar County, Tex., because of the existence of hog cholera. This action is deemed necessary to prevent further spread of the disease. The restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will apply to the quarantined portions of such county.

The amendments also exclude a portion of Smith County, Tex., from the areas quarantined because of hog cholera. No areas in Smith County, Tex., remain under the quarantine. Therefore, the restrictions pertaining to the interstate movement of swine and swine products from or through quarantined areas as contained in 9 CFR Part 76, as amended, will not comply to the excluded area, but will continue to apply to the quarantined areas described in § 76.2(e). Further, the restrictions pertaining to the interstate movement of swine and swine products from nonquarantined areas contained in said Part 76 will apply to the area excluded from quarantine.

Insofar as the amendments impose certain further restrictions necessary to prevent the interstate spread of hog cholera, they must be made effective immediately to accomplish their purpose in the public interest. Insofar as they relieve restrictions, they should be made effective promptly in order to be of maximum benefit to affected persons.

Accordingly, under the administrative procedure provisions in 5 U.S.C. 553, it is found upon good cause that notice and other public procedure with respect to the amendments are impracticable, unnecessary, and contrary to the public interest, and good cause is found for making them effective less than 30 days after publication in the FEDERAL REGISTER.

Done at Washington, D.C., this 16th day of February 1971.

F. J. MULHERN,
Acting Administrator,
Agricultural Research Service.

[FR Doc.71-2339 Filed 2-19-71;8:46 am]

Title 10—ATOMIC ENERGY

Chapter I—Atomic Energy Commission

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

General Design Criteria for Nuclear Power Plants

The Atomic Energy Commission has adopted an amendment to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which adds an Appendix A, "General Design Criteria for Nuclear Power Plants."

Section 50.34(a) of Part 50 requires that each application for a construction permit include the preliminary design of the facility. The following information is specified for inclusion as part of the preliminary design of the facility:

(i) The principal design criteria for the facility

(ii) The design bases and the relation of the design bases to the principal design criteria

(iii) Information relative to materials of construction, general arrangement, and the approximate dimensions, sufficient to provide reasonable assurance that the final design will conform to the design bases with adequate margin for safety.

The "General Design Criteria for Nuclear Power Plants" added as Appendix A to Part 50 establish the minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. They also provide guidance in establishing the principal design criteria for other types of nuclear power plants. Principal design criteria established by an applicant and accepted by the Commission will be incorporated by reference in the construction permit. In considering the issuance of an operating license under Part 50, the Commission will require assurance that these criteria have been satisfied in the detailed design and construction of the facility and that any changes in such criteria are justified.

A proposed Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" to 10 CFR Part 50 was published in the FEDERAL REGISTER (32 F.R. 10213) on July 11, 1967. The comments and suggestions received in response to the notice of proposed rule making and subsequent developments in the technology and in the licensing process have been considered in developing the revised criteria which follow.

The revised criteria establish minimum requirements for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission, whereas the previously proposed criteria would have provided guidance for applicants for construction permits for all types of nuclear power plants. The revised criteria have been reduced to 55 in number, include definitions of important terms, and have been rearranged to increase their usefulness in the licensing process. Additional criteria describing specific requirements on matters covered in more general terms in the previously proposed criteria have been added to the criteria. The Categories A and B used to characterize the amount of information needed in Safety Analysis Reports concerning each criterion have been deleted since additional guidance on the amount and detail of information required to be submitted by applicants for facility licenses at the construction permit stage is now included in § 50.34 of Part 50. The term "engineered safety features" has been eliminated from the revised criteria and the requirements for "engineered safety features" incorporated in the criteria for individual systems.

Further revisions of these General Design Criteria are to be expected. In the course of the development of the revised criteria, important safety considerations were identified, but specific requirements related to some of these considerations have not as yet been sufficiently developed and uniformly applied in the licensing process to warrant their inclusion in the criteria at this time. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(i) Consideration of the need to design against single failures of passive components in fluid systems important to safety.

(ii) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem and the required interconnection and independence of the subsystems have not yet been developed or defined.

(iii) Consideration of the type, size, and orientation of possible breaks in the components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

(iv) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of the protection systems and reactivity control systems.

In addition, the Commission is giving consideration to the need for development of criteria relating to protection against industrial sabotage and protection against common mode failures in systems, other than the protection and reactivity control systems, that are important to safety and have extremely high reliability requirements.

It is expected that these criteria will be augmented or changed when specific requirements related to these and other considerations are suitably identified and developed.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification to be effective 90 days after publication in the FEDERAL REGISTER. The Commission invites all interested persons who desire to submit written comments or suggestions in connection with the amendment to send them to the Secretary, U.S. Atomic Energy Commission, Washington, D.C. 20545, Attention: Chief, Public Proceedings Branch, within 45 days after publication of this notice in the FEDERAL REGISTER. Such submissions will be given consideration with the view to possible further amendments. Copies of comments may be examined in the Commission's Public Document Room at 1717 H Street NW., Washington, DC.

1. Section 50.34(a)(3)(i) is amended to read as follows:

§ 50.34 Contents of applications; technical information.

(a) *Preliminary safety analysis report.* Each application for a construction permit shall include a preliminary safety analysis report. The minimum information to be included shall consist of the following:

* * * * *

(3) The preliminary design of the facility including:

(i) The principal design criteria for the facility.[3] Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants for construction permits in establishing principal design criteria for other types of nuclear power units:

* * * * *

2. A new Appendix A is added to read as follows:

APPENDIX A—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

Table of Contents

[3] General design criteria for chemical processing facilities are being developed.

INTRODUCTION

Pursuant to the provisions of § 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, some of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined. Their omission does not relieve any applicant from considering these matters in the design of a specific facility and satisfying the necessary safety requirements. These matters include:

(1) Consideration of the need to design against single failures of passive components in fluid systems important to safety. (See Definition of Single Failure.)

(2) Consideration of redundancy and diversity requirements for fluid systems important to safety. A "system" could consist of a number of subsystems each of which is separately capable of performing the specified system safety function. The minimum acceptable redundancy and diversity of subsystems and components within a subsystem, and the required interconnection and independence of the subsystems have not yet been developed or defined. (See Criteria 34, 35, 38, 41, and 44.)

(3) Consideration of the type, size, and orientation of possible breaks in components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss-of-coolant accidents. (See Definition of Loss of Coolant Accidents.)

(4) Consideration of the possibility of systematic, nonrandom, concurrent failures of redundant elements in the design of protection systems and reactivity control systems. (See Criteria 22, 24, 26, and 29.)

It is expected that the criteria will be augmented and changed from time to time as important new requirements for these and other features are developed.

There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

DEFINITIONS AND EXPLANATIONS

Nuclear power unit. A nuclear power unit means a nuclear power reactor and associated equipment necessary for electrical power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Loss of coolant accidents. Loss of coolant accidents mean those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.[1]

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[2]

Anticipated operational occurrences. Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

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[1] Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

[2] Single failures of passive components in electrical systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

CRITERIA

I. Overall Requirements

Criterion 1—Quality standards and records. Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Criterion 2—Design bases for protection against natural phenomena. Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3—Fire protection. Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Criterion 4—Environmental and missile design bases. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the envrionmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Criterion 5—Sharing of structures, systems, and components. Structures, systems, and components important to safety shall not be shared between nuclear power units unless it is shown that their ability to perform their safety functions is not significantly impaired by the sharing.

II. Protection by Multiple Fission Product Barriers

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

Criterion 11—Reactor inherent protection. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Criterion 12—Suppression of reactor power oscillations. The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Criterion 13—Instrumentation and control. Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Criterion 14—Reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 16—Containment design. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 17—Electrical power systems. An onsite electrical power system and an offsite electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electrical power sources, including the batteries, and the onsite electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights of way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power sources and the other offsite electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electrical power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electrical power sources.

Criterion 18—Inspection and testing of electrical power systems. Electrical power systems important to safety shall be designed to permit periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Criterion 19—Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

III. Protection and Reactivity Control Systems

Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Criterion 25—Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods or unplanned dilution of soluble poison.

Criterion 26—Reactivity control system redundancy and capability. Two independent reactivity control systems of different design principles and preferably including a positive mechanical means for inserting control rods, shall be provided. Each system shall have the capability to control the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operations, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Criterion 27—Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29—Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

IV. Fluid Systems

Criterion 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Criterion 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Criterion 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Criterion 33—Reactor coolant makeup. A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Criterion 34—Residual heat removal. A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 35—Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of coolant accident at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 36—Inspection of emergency core cooling system. The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Criterion 37—Testing of emergency core cooling system. The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 38—Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 39—Inspection of containment heat removal system. The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Criterion 40—Testing of containment heat removal system. The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Criterion 41—Containment atmosphere cleanup. Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Criterion 42—Inspection of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Criterion 43—Testing of containment atmosphere cleanup systems. The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Criterion 44—Cooling water. A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Criterion 45—Inspection of cooling water system. The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Criterion 46—Testing of cooling water system. The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

V. Reactor Containment

Criterion 50—Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51—Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Criterion 52—Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Criterion 53—Provisions for containment testing and inspection. The reactor containment shall be designed to permit (1) inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Criterion 54—Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55—Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Criterion 56—Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57—Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

VI. Fuel and Radioactivity Control

Criterion 60—Control of releases of radioactive materials to the environment. The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Criterion 61—Fuel storage and handling and radioactivity control. The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Criterion 62—Prevention of criticality in fuel storage and handling. Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Criterion 63—Monitoring fuel and waste storage. Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Criterion 64—Monitoring radioactivity releases. Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

(Secs. 161, 182, 68 Stat. 948, 953; 42 U.S.C. 2201, 2232)

Dated at Washington, D.C., this 10th day of February 1971.

For the Atomic Energy Commission.

W. B. McCool,
Secretary of the Commission.

[FR Doc.71-2370 Filed 2-19-71;8:48 am]

Title 14—AERONAUTICS AND SPACE

Chapter I—Federal Aviation Administration, Department of Transportation

[Docket No. 71-EA-13; Amdt. 39-1155]

PART 39—AIRWORTHINESS DIRECTIVES

American Aviation Corp.

The Federal Aviation Administration is amending § 39.13 of Part 39 of the Federal Aviation Regulations so as to issue an airworthiness directive applicable to

DOCKET NUMBER
PROPOSED RULE 50

OAK RIDGE NATIONAL LABORATORY

OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

September 6, 1967

Mr. H. L. Price
Director of Regulation
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Price:

Subject: Review of USAEC "General Design Criteria for Nuclear Power Plant Construction Permits" Federal Register, July 11, 1967

The subject document has been reviewed by members of the staff of the Nuclear Safety Information Center. We realize and appreciate the great amount of work that your staff has done in bringing these criteria to their present form. We participated in the initial review of the criteria when they were issued in November 1965 and we are pleased to have the opportunity to review this later version. Our comments are enclosed in two parts: (1) general comments which apply to the entire set of criteria and (2) specific comments on the individual criteria and in a few cases on sections such as VII, Engineered Safety Features.

With a few exceptions, the scope of the criteria seems broad enough and generally well organized. We do have rather extensive comments on those criteria which deal with protection systems. A difficult problem is that of assessing reliability. The "single failure criterion" is an attempt to relieve this situation, but its application is subjective and it has different meanings to different individuals. Another problem area is that of the use of the same instruments for both operating the plant and providing protection. We believe that such interdependence can only degrade the reliability and performance of the protection system. Problems such as these make the task of writing criteria and standards quite difficult.

Further, the absence of clear definitions of terms, which to many are rather loosely understood, could limit the effectiveness of the criteria. We feel that there is a critical need for these definitions.

We again wish to commend you for the significant contribution represented by these criteria. If you have questions concerning our comments, we will be glad to discuss them with you.

Sincerely yours,

Wm. B. Cottrell

Wm. B. Cottrell, Director
Nuclear Safety Information Center

WBC:JRB:jt

Enclosure

cc A. J. Pressesky

General Comments

1. The ramifications of civil disobedience, riots, strikes, sabotage, and the like have not even been mentioned. With this vast potential risk in mind, should not the physical security of the plant be considered?

2. Since these criteria will be used by many groups whose terminology is not always (or even usually) in agreement, a set of definitions is badly needed. For example - what is a system, component, engineered safety feature, failure, redundancy, channel, surveillance, monitoring, malfunction, protection system, loss of coolant accident, etc.?

3. Since "single failure criteria" are to be applied to systems other than those for control (for which criterion 21 is the definition), it is extremely important that they be clearly defined for all systems.

4. Since the introduction uses the phrase "nuclear reactor plant" why is the phrase "reactor facility" used in the text of several of the criteria to mean the same thing?

Specific Comments

Title - General Design Criteria for Nuclear Power Plant Construction Permits

The title is really not grammatically correct, since it infers that we are designing a "construction permit".

Criterion 2 - Performance Standards

1. Line 7: Delete "performance" since this could be construed as applying to operating performance only.

2. In regard to earthquakes the "appropriate margin for withstanding forces greater than those recorded . . ." has not been defined here and furthermore it would be extremely difficult to do so at least with our present understanding of earthquake phenomena. Therefore, the criterion should state what constitutes an adequate margin.

Criterion 4 - Sharing of Systems

We agree with criterion 4 as it applies to the nuclear reactor plant but it should be extended to apply to systems, sub-systems, and especially engineered safety features.

Criterion 5 - Records Requirements

1. Line 2: Should read, "Records of the design, fabrication, inspection, testing and construction of . . ." to be sufficiently inclusive. The performance of engineered safety features must be determined as a datum for evaluation of subsequent tests required of the system. For example, criterion 46 states that active components be periodically tested for required performance.

2. Line 5: Change "its" to "his" to refer to the operator's control.

Criterion 8 - Overall Power Coefficient

For this entire criterion it might be better to say that "the reactor shall be designed so that either the overall power coefficient in the power operating range shall not be positive or reliable controls which will eliminate or minimize the undesirable effects of a positive power coefficient shall be provided, tested and proved effective."

Criterion 10 - Containment

We infer from subsequent criteria that the protection system is not considered an engineered safety feature even though there are reactors that depend upon the protection systems to work in order not to overstress the containment. Thus, either "engineered safety features" should be defined to include the reactor protective system, i.e., scram functions, or this and other functions should be specifically mentioned. We prefer the former alternative.

Criterion 11 - Control Room

The aims of this criterion are certainly desirable but it is difficult if not impossible to prove the criterion has been met. However, some clarification is needed, for example, if a fire in a panel renders the controls of some emergency system inoperable, the criterion can be interpreted to mean that two separate control rooms are required. Is this the intent?

Criterion 13 - Fission Process Monitors and Controls

1. Line 4: Delete "throughout core life and" since it is redundant.

2. The examples cited should either be deleted or augmented by a more comprehensive set including flux, hot spots, etc.

Criteria 14 and 15 - Core Protection Systems and Engineered Safety Features

These criteria exemplify the fact that a more detailed definition of containment and engineered safety features needs to be included. One could define the engineered safety features as including scram system, core protection system, etc., and then eliminate Criterion 14.

Suggested Criterion - Monitoring Engineered Safety Features

We suggest that this criterion be inserted at this point: Instrumentation shall be provided to monitor the performance of engineered safety features during the course of the accident and to monitor the condition of the reactor itself under these conditions.

Criterion 16 - Monitoring Reactor Coolant Pressure Boundary

This criterion defines the monitoring that is necessary to prove compliance with Criterion 9. (Similar proof is required by Criterion 36) In cases of this nature cross referencing of criteria should be made for the sake of clarity.

Criterion 17 - Monitoring Radioactivity Releases

This criterion was written to specify monitoring to meet the specifications of Criterion 70, which should be cross referenced here.

Criterion 18 - Monitoring Fuel and Waste Storage

Specification of criticality monitoring should be included in this criterion; for example, as by reference to 10 CFR, Part 70.34.

Criterion 19 - Protection Systems Reliability

There is no guide for determining whether or not the functional reliability and in-service testability is commensurate with the safety functions to be performed. Every designer could claim that his system met this criterion, and challenge a reviewer to show otherwise. Arguments about this criterion most likely will include comparisons to somewhat similar protection systems for somewhat similar nuclear power plants that have been reviewed and approved.

This criterion is of questionable value and we recommend its omission. A set of rules for designing protection systems would be more useful than a general statement of desirable results.

Criterion 20 - Protection Systems Redundancy and Independence

The criterion is not clear as to the extent of the effects of a single failure that need consideration. Apparently, considerations of effect are to be limited to a component or channel - resulting in a severe limitation in the value of this criterion. This is another example of a criterion where definitions are needed; for example, component, channel, and system need to be defined.

Criterion 21 - Single Failure Definition

A judgment of the extent of failures caused by a single event hinges on credibility. First, there is the probability of the initiating event, then the probability of progressive failures. A single event of sufficient magnitude will certainly prevent the functioning of the protection system. Detailed guidelines for describing the required independence of redundant equipment are needed. Examples are spacing between cables carrying redundant signals, methods of separating electronic equipment handling redundant signals, methods of isolating redundant logic devices which combine redundant signals, etc. Unless more detailed information is given as to what is to be considered credible, this criterion serves little purpose.

Criterion 22 - Separation of Protection and Control Instrumentation Systems

This criterion apparently recognizes the need for separating protective and control instrumentation but compromises this objective with the qualifications permitted. The net effect is to permit the intimate intermingling of the system that normally operates the plant and the system that is intended to afford protection. We strongly recommend that no exceptions be permitted to the separation of these two systems as the only effective means to insure the vital integrity of the protection system.

Both of these systems in the new and larger reactors are complex. Despite the use of buffer amplifiers in attempting to isolate the effects of failures in the two systems, the systems are not independent when the same signals are coupled into each. Additionally, the objectives of operation are not those of protection. When the two systems are intermingled, signal processing equipment is invariably designed for operating the plant rather than for protection. Inadequate control demands that corrections must be made in the equipment to allow operation, but inadequate protection equipment may be discovered only after their need during an accident. Mixing of the two systems as allowed by this criterion diverts design attention from the requirements of protection to those of operation. Such mixing also increases the probability that protection will be lost as the result of a failure in the control system that initiates the accident requiring protection.

The basic justification for independence of protection and operation systems, in our opinion, is the relative ease with which the protection function can be assured with independence, and the great difficulty of realizing such assurance with interdependence. We believe it is easier to separate the systems than to assure that their interactions are harmless. We believe it is easier to maintain independence than to insure, for the lifetime of the plant, that deliberate changes or inadvertent alteration of the operation system will not adversely affect the protection function.

The dismal list of accidents caused by design errors, and the much larger list of design errors caught before they caused accidents, lead us to believe that design errors will continue to occur. We believe further that independence of operation and protection is one of the best defenses against the possibility that a design error may cause an unprotected accident.

It may be possible that for some combinations of protection and operation instruments no conceivable failure of the operation function involved can result in a situation requiring action of the protection function involved. To the extent that this can be proved, both initially and throughout reactor lifetime, the particular interdependence could be acceptable. A hypothetical example is the instrumentation used to measure and control the pressure of a sealed containment enclosure. The operation function is used principally to provide a pressure differential between the inside of the containment and the outside, and thus to provide a means for surveillance of the leakage rate.

The protection function might be to initiate reactor shutdown, emergency cooling, and isolation of process piping if a rise in containment pressure should indicate the presence of a serious leak of potentially radioactive fluids. It might be demonstrable that no failure whatever of this instrumentation could induce a substantial leak of radioactive fluid, in which case no real interdependence of operation system and protection system would in fact exist.

The basis of the above example is the impossibility that failure of the operational function or equipment could ever, under any circumstances, lead to a situation where the protection function would be needed. Therefore, sharing of equipment (common elements) between the protection system and the operation system could not lead to interaction between the two systems. It is difficult to prove conclusively this lack of functional interaction. More difficult is the problem of ensuring that this lack of interaction can and will be maintained throughout the life of the plant. Operators are not designers; operators in charge of the plant at the end of its 40-year life are not the ones who may have discussed protection problems with the designers at the beginning. Subtle considerations are apt to be forgotten or ignored. It is easy to forget that plant protection was originally based on the impossibility that failure of certain operation instruments could result in a need for protection-system function.

<u>Criterion 24</u> - Emergency Power for Protection Systems

Design requirements related to power supply include consideration of both Criteria 24 and 26. There is an anomaly here in that Criterion 24 permits the protection system to require power to provide protection, whereas Criterion 26 requires the system to fall into a safe or tolerable state on loss of power. To the extent that Criterion 26 can be met, alternate power sources become an economic or operational consideration rather than being needed for safety.

<u>Criterion 25</u> - Demonstration of Functional Operability of Protection Systems

We agree with the intent of this criterion but suggest that the wording be changed to state ". . . demonstrate that no failure causing a reduction of redundancy . . ." rather than ". . . demonstrate that no failure or loss of redundancy . . .". Some systems may have extra elements whose failures do not reduce the redundancy claimed for the system.

<u>Criterion 26</u> - Protection Systems Fail-Safe Design

This criterion places a requirement not only on the protection system but on the plant as well. For example, a plant design could be such that operation of the protection mechanism when not needed would be highly undesirable. (An illustration is the closure of the steam stop valves in a

BWR.) Criterion 26 requires the plant to be able to accept operation of the protection system when not needed. We believe this is a good objective and we support this criterion.

Section V - Reactivity Control

1. The title of this section should be "Reactivity Control for Reactor Shutdown".

2. This group of criteria should distinquish more clearly between functions of reactivity control; namely, the dynamic reactivity reduction process and the static holddown functions. The first function must be performed at such times as in power transients and loss-of-coolant accidents with the objective of preventing exceeding "acceptable fuel damage limits" referred to in Criteria 28 and 29. Margins expressed in terms of shutdown parameters are inappropriate and inadequate for the dynamic function.

   The reliability with which each function must be carried out depends upon the seriousness of the consequences of failure of that function.

Criterion 27 - Redundancy of Reactivity Control

This criterion is not clear. It does not state whether the two reactivity control systems (1) should both be capable of both increasing and decreasing reactivity for operation, or (2) should both be capable of fast shutdown, or (3) should one be for fast shutdown and one for holddown. We recommend that the word "shutdown" be substituted for "control" in this criterion. These systems should also meet the requirements of Criteria 28, 29, 30, 31, and 32.

Criteria 28, 29, and 30 taken together indicate that one of the shutdown systems is not required to cope with positive transients and is essentially a method of obtaining reactivity holddown capability. However, reactors that must be shut down rapidly to allow the containment system to function need two separate and fast shutdown systems. A single fast or "primary" shutdown system together with a "holddown", or slow, "secondary" shutdown system is not satisfactory in this case.

Criterion 29 - Reactivity Shutdown Capability

As stated in our comments on Criterion 27, some reactors require a shutdown to allow the containment to function. In such cases, this criterion should require that two shutdown systems be applied. Each such system should be capable of preventing an unacceptable situation.

This criterion carries a reference to shutdown margin that could well be made a separate criterion as the shutdown requirements are a function of the number of rods, reactor operating conditions and function desired (e.g., reduction of nuclear power level or holddown of the subcritical reactor). Although we have not addressed ourselves to these conditions in detail, we believe that a margin much greater than the worth of the most effective control rod is needed for reactors having many rods.

Criterion 30 - Reactivity Holddown Capability

In cases requiring the reactor to be shut down in order to achieve containment, two of these systems should be required. See comments on Criteria 27 and 29.

Criterion 31 - Reactivity Control Systems Malfunction

This criterion should be expanded to include all failures of the plant operating system that are capable of increasing reactivity. In particular this criterion should not be limited to the unplanned withdrawal of only one control rod since a failure of the control rod operating system may not be restricted to the withdrawal of only one rod. All failures that may affect the performance of the control rod operating system must be considered. Of a more general nature, all failures that can introduce reactivity increases must be considered. In addition to control rods, there are coolant temperature changes, and perhaps even void effects that need analysis.

Criterion 33 - Reactor Coolant Pressure Boundary Capability

We agree with the intent of the criterion but it is not clear what is meant by "positive mechanical means" for preventing a rod ejection. A definition is needed.

Section VII - Engineered Safety Features

With the exception of reactor shutdown systems, all other engineered safety features are discussed in this section. These are: emergency power system, emergency core cooling system, containment enclosure system, containment pressure-reducing system (including containment heat removal), and air cleaning systems.

For each of these systems, there should be criteria for design of the system and their components as well as criteria for testing and inspection.

The objective of these criteria would be clearer if each system were treated in separate subsections and the criteria for each were set up in parallel form. Thus, there would be criteria for the inspection and testing of emergency power system (now covered in only Criterion 39) as well as the inspection and testing criteria for the other engineered safety features. Criterion 52, "Containment Heat Removal Systems," would be grouped with Criteria 58-61 with which it is generally associated. Such a rearrangement raises questions on other points of apparent inconsistancy, e.g., Criterion 60 is seen to be but a special case of Criterion 61, etc.

Criterion 37 - Engineered Safety Features Basis for Design

Again a definition of engineered safety features is necessary. For example, if the scram must work in order that the containment not be overstressed, then the scram system must be considered part of an engineered safety feature.

Criterion 38 - Reliability and Testability of Engineered Safety Features

We agree with this criterion. However, its title and inclusion in Section VII, both of which pertain only to engineered safety features, does not reflect its more general applications which include "inherent" as well as "engineered safety features". It would more appropriately be included in Section I.

Criterion 39 - Emergency Power for Engineered Safety Features

A difficult point in the application of this criterion is that of redundancy in the offsite power system. For example, a plant failure that results in shutting off the electric generator driven by the reactor could produce the loss of all offsite power. The probability of this consequential loss of offsite power varies widely as a result of changes in the power system and of variations in power system load. As a result of this wide variation in the reliability of offsite power, we recommend that this criterion require that redundant and independent onsite power system be required such that onsite power alone be capable of supplying the needs of the engineered safety features after a failure of a single active component in the onsite power system. We do not believe that the offsite power is really independent of the power from a main generator operated from the reactor to be safeguarded.

Criterion 40 - Missile Protection

Analysis shall be made to show that fragments and components that could be ejected from highly pressurized system's rotating equipment would not impair the function of an engineered safety feature. Typical missiles requiring analyses are such items as primary system valves, flanges, instrumentation, etc. When rotating equipment is not completely contained, such as in a concrete vault, a missile map should be provided for rotating equipment (e.g., main turbines, pumps, etc.)

Criterion 41 - Engineered Safety Features Performance Capability

We agree with this criterion as far as it goes. In particular the detailed requirements for the emergency core cooling system as contained in Criterion 44 illustrate the desired amplification (but for that system only). Thus, it could be generalized and added to Criterion 41 as follows: "The performance of each engineered safety feature shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident."

Criterion 42 - Engineered Safety Features Components Capability

We see no need to limit this criterion to the loss-of-coolant accident and suggest that . . . "by the effects of a loss-of-coolant accident" be changed to read "the effects of the accident for which the function is required."

Criterion 43 - Accident Aggravation Prevention

It is not obvious what purpose this criterion is intended to serve. If something specific is in mind here it should be stated, i.e., are we worried about the core becoming critical again, or inducing a thermal shock, etc. Perhaps this should not even appear here but be in the general discussion.

Criterion 44 - Emergency Core Cooling Systems Capability

As noted in the discussion on Criterion 41, we would restrict this criterion to the first two sentences (having already included the remainder of this criterion as a general requirement in Criterion 41). However, as we interpret the intent of these sentences, each of the two emergency cooling systems should cover the whole range of pipe break conditions up to the maximum. To make this point clearer, it might be better to rephrase the second sentence defining the cooling system requirements as follows: "For each size break in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe, at least two emergency core cooling systems, preferably of different design principles and each with a capability for accomplishing abundant emergency core cooling, shall be provided."

Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems

We agree with the intent of this criterion and suggest that in addition to "the transfer to alternate power sources" the operation of the reactivity control system (which must shutdown the reactor and then provide holddown in the cold condition after the loss-of-coolant accident) should be mentioned.

Criterion 49 - Containment Design Basis

We agree with the intent of this criterion but feel that the following need some elaboration:

Line 10: "Considerable Margin" should be defined in some manner.

Line 13: What degree of failure of the emergency core cooling system is assumed?

Criterion 50 - NDT Requirement for Containment Material

This criteria needs further clarification. The temperature of the steel members in question under normal operating and testing conditions should be defined, i.e., the temperature of the component when the ambient temperature is at its lowest recorded (or perhaps expected) value. Furthermore, the requirement of NDT + 30° F has no meaning in the eyes of the stress analyst although it has found some usage. This temperature is half way between NDT and FTE and unless there is adequate justification of which we are unaware, we recommend using NDT + 60° F which defines the transition, e.g., temperature at which cracks won't propagate at stresses less than yield.

Criterion 51 - Reactor Coolant Pressure Boundary Outside Containment

The intent of this criterion is not clear. It would appear that Criterion 53 which requires redundant valving would also cover reactor containment coolant boundaries outside containment. If, however, it is intended to require extensions of the containment, it should be specifically stated. In any event . . . delete "appropriate" and "as necessary" in lines 4 and 5 and the entire last sentence which begins, "Determination of . . .". These words do not materially contribute to the sense of the statement of the criterion and therefore should be omitted.

Criteria 54, 55, and 56 - Containment Leakage Rate Testing, Containment Periodic Leakage Rate Testing, and Provisions for Testing of Penetrations

Following the words "design pressure" it is suggested that "defined by Criterion 49" be inserted.

Criterion 56

This criterion is not sufficiently inclusive. The types of penetrations which should be tested should NOT be limited to the two that are mentioned, but for instance should also include electrical penetrations and piping penetrations that do not require expansion joints. The penetration testing is usually done at greater than design pressure.

Criterion 66 - Prevention of Fuel Storage Criticality

We do not understand the implication of "or processes" at the end of the first sentence, nor do we believe that it is practical to depend upon procedural controls to prevent accidental criticality in storage facilities of power reactors. Hence, the last sentence of this criterion should be changed to read as follows: "Such means as geometrically safe configuations shall be used to insure that criticality cannot occur."

Criterion 67 - Fuel and Waste Storage Decay Heat

To the extent that removal of decay heat is a function necessary to prevent escape of fission products, decay heat removal systems should be designed to the same requirements for redundancy, inspectability, and testability as engineered safety features on reactors. This should include facilities for supplying additional coolant fluid in the event of accidental loss.