EXHIBIT 26

Transcript, Deposition of Anthony C. Attard, Ph.D., May 11, 2000

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	:	Docket No. 50-423-LA-3
Northeast Nuclear Energy Company	:	
(Millstone Nuclear Power Station, Unit No. 3	:	MAY 11, 2000

DEPOSITION OF DR. ANTHONY C. ATTARD

CERTIFIED
COPY

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A P P E A R A N C E S :

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For Connecticut Coalition Against Millstone
Long Island Coalition Against Millstone
The Intervenors

WINSTON & STRAWN
1400 L Street, N.W.
Washington, D.C. 20005-3502
BY: DAVID A. REPKA, ESQ. and
DONALD P. FERRARO, ESQ.

For Northeast Nuclear Energy Company

NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555
BY: Ann P. Hodgdon, NRC Staff Counsel

ALSO PRESENT:

David W. Dodson
Laurence T. Kopp, Ph.D.
David Lochbaum
Victor Nerses
Gordon Thompson, Ph.D.

INDEX OF EXAMINATION

INDEX OF EXHIBITS

(None offered at this deposition)

* * * * *

Deposition of DR. ANTHONY C. ATTARD, a witness in the above-entitled action, taken at the request of the Intervenors pursuant to 10 CFR Section 2.740a before Kathryn Orofino, a Notary Public within and for the State of Connecticut, at the Mystic-Noank Library, 40 Library Street, Mystic, Connecticut, commencing at 12:20 p.m.

* * * * *

STIPULATIONS

The deposition is to be used for discovery or as evidence in this proceeding only; objections or motions to strike will not be considered to be waived except as to matters of form; the Deponent will be given a right to read and sign the transcript when it is complete; the original of the transcript will be forwarded to the deposing attorney who will provide the opportunity for the witness to read and sign; and the original will be filed with the Commission in accordance with the Commission's rule of 10 CFR part 2.

D R . A N T H O N Y C . A T T A R D ,

of United States Nuclear Regulatory Commission, Washington, D.C., 20555, a witness in the above-entitled action, having been duly sworn by Kathryn Orofino, a Notary Public within and for the State of Connecticut, was examined and testified on his oath as follows:

* * * * *

THE REPORTER: Are you going to have the same stipulations?

MR. REPKA: That's fine.

MS. BURTON: You can just incorporate them.

EXAMINATION BY MS. BURTON

Q Dr. Attard, first of all, am I pronouncing that properly?

A Oh, yes. Yes.

Q You have been a nuclear reactor physicist with the NRC since 1990; is that correct?

A Uh-huh.

Q Can you tell us what role you have played in these proceedings concerning the application by the licensee to rerack in Unit 3 of the spent fuel pool?

A Yes, I was given the job of reviewing the reracking submittal, the amendment request, and --

which I had started back in -- last year, actually.

Q Do you recall when last year?

A I would say around the June time frame.

And around September, maybe a little later, this situation of the contentions started to -- at the time I heard about it from the PM, who was not Victor at the time, it was somebody else.

And so I put it aside for a little bit while -- to see what the outcome would be from the contentions and so forth. In the meantime, we're always working on several things at once. This is just one amendment that one has on their desk, so to speak.

Q So in other words, you were assigned to analyze the amendment request in order to aid --

A Review. Review it.

Q -- the Commission in deciding whether to grant it or deny it or whatever?

A Right.

Q And in addition, you have been asked to participate in these proceedings?

A That's correct.

Q Is it part of your assignment to continue to provide the NRC staff with the benefit of your insight as this process continues?

A Yes, on both -- both -- you know, the

amendment and the contention aspect of it, too.

Q Did you participate in the formulation of the finding of a no significant impact?

A I don't know. Not yet anyway. The review is still ongoing, so if it's part of the package, I would, you know, eventually see it, so --

Q Before these proceedings, had you ever participated in providing input in adjudicatory proceedings --

A No.

Q -- licenses, challenges?

A No.

Q Now, you have put the date at June '99 approximately when you were first involved in this matter.

Can you please tell us how you went about your analysis; what you were requested to do.

A Yes. Usually what I do, as I do with all amendments pretty much, is I read through the package, you know, and go looking for what I call red flags that may pop up out of the pages, you know. And then quite often you talk with your colleagues, you know, to see whether something similar has been done before, see if there's a precedence to it, you know. And you start collecting any info you may need for it, whether it's

Reg. Guides or 10CFR50.

And so I try -- usually I try to take a quick cut off any REI's that I may want to address as a first cut. I usually do this a couple of times just to make sure that there is no major show stopper, sometimes people refer to them, in the package, including the whole package.

And if there isn't, it's -- you know, I start to look into it a little deeper. There's almost two or three levels that I go into.

Again, you interact -- well, I did anyway -- seeked Larry's comments on things, because I --

Q Who is Larry?

A Dr. Kopp, excuse me. Dr. Kopp.

Q Uh-huh.

A -- comments, because he has been in the spent fuel pool area for a great number of years. So -- and I'm still in that process now.

Q So the spent fuel pool is not one of your particular areas of specialty?

A Not really, no. I -- I inherited it, so to speak, because there are -- people leave. They would like to have some type of continuity, you know, and I was picked to do that.

Q Now, you looked at first for red flags. Did you see any?

A The -- the 800 ppm part with regard to being present, not so much the ppm concentration, but rather the while moving fuel was one that I was going to -- I was going to pursue for sure, you know, either through a conference call or perhaps some other means.

Q And why is that?

A I wasn't sure of what -- what it meant. It's how I -- how I was interpreting it was that -- and this was before I went into the tech. specs. I had to look to see about the 2600 ppm. So this was the first -- like I said, the first sitting down and reading, okay, was to make sure that I was understanding it correctly.

I read it as meaning that there would not be or they would not be concerned about having concentration in the pool, which didn't make any sense at all, so I knew I was wrong, but I wanted to, obviously, follow up on it.

And so -- which I did eventually. We sent a set of REI's out, actually. And in the meantime, I came in contact with this one incident where I did talk to Dr. Kopp regarding chemistry procedures that I'm sure they were in existence. And actually, I knew that they would be monitoring at some interval, which I did not know at the time. So it was still -- it was still in the very preliminary stage.

Q Prior to this application had you ever reviewed another amendment concerning a spent fuel pool reracking?

A No. No.

Q Well, you mentioned the red flag being the 800 parts per million?

A Uh-huh.

Q Why was that a red flag to you? I'm not sure I understood.

A Oh, not so much the 800. Perhaps I should clarify that. It's the while moving fuel is the part that kind of -- I wanted to -- I was going to seek clarification on. Not so much the 800, but while moving fuel is the part that gave me a little bit of heartburn.

Q And tell me why. What about it gave you heartburn?

A Only because like I said, I wasn't sure what that meant. I knew that I'm misinterpreting it about what they were asking or what they are requesting, so I had to -- I made it a point to first go and talk to Larry -- Dr. Kopp, and then eventually, you know, we made up a set of REI's, two or three REI's -- I don't remember how many at the time -- after we had a conference call with the licensee to pursue that.

Q So now do you recognize the need to maintain boron at the spent fuel pool at the level that is proposed?

A At the 800 level or -- right now it's still --

Q At the 800 level?

A I -- no, because their tech. specs say they have to be within -- they have to be greater or equal to 2600 ppm, so really the 800 is just to show that if they had a misplacement or, you know -- it was -- again, it fell into that double contingency situation.

Q Were there any other red flags?

A No.

Q Now, when you set about discharging your assignment to analyze the application, can you please tell us what standards you had to meet --

A Oh, by that --

Q -- or the application had to meet.

A Well, again, there was -- it was kind of done for me in a way.

Q It was what?

A I was kind of presented in the front of the Holltech (ph) report. For example, you know, they list a series of -- I think it's Rec. Guide 11 -- 1.13, the so-called Grimes letters, Grimes letter, Dr. Kopp's letter. There's about five or six bullets in there that talks about the -- what they had to meet.

And I would have went to the same thing. I mean, I would have resorted to the same -- same documents, if you like.

Q So in other words, the report that was submitted by Holltech set out the standards that it believed --

A That's correct.

Q -- it had to meet?

A Yes.

Q And you believe that that assessment was correct?

A Yeah, again, you know, I would have checked with Dr. Kopp, for example, to see if there was anything else that was either left out or, you know.

Q So again, going through those, that was the Reg. Guide and --

A 1.13, I believe, yeah.

Q And Dr. Kopp's memo?

A Dr. Kopp's memo, and I think there was --

Q And what else?

A I think it was Brian Grimes.

Q The Grimes letter?

A The Grimes letter, yeah.
Q Anything else?
A I think the ANSI so and so.
Q The ANSI what?
A I forget the number of it offhand, but it's the same ANSI that talks about the double contingency principal that's stated in there.
Q And anything else?
A No, not that I know of.
Q So are you satisfied that those four components constitute all of the standards that the NRC applies to consider license amendment to rerack in the spent fuel pools?
A I think they go a long way to help whoever is doing the review to do a satisfactory job. Satisfactory meaning, of course, that all the safety requirements are met and the criticality, in particular, is the .95.
Q Have you determined that this application meets all those so-called standards?
A I have not yet.
Q What else do you need to look at?
A Well, I mean, I haven't looked at all of them is what I meant.
Q Haven't looked at what?

A At all those ones that I told you. I haven't individually went down the list and looked at all of those yet.

Q You have never looked at some of these?

A I have. In terms of not necessarily in that order or -- or checked off every one as I went down. In other words, I'm still in the process of doing the review is what I'm saying.

Q Prior to this assignment have you had occasion to look at Dr. Kopp's memo?

A No, not after this, no.

Q What about the Grimes letter?

A No. None of these --

Q The particular ANSI standards that you referred to?

A No. I've heard about them thrown around, you know, discussed, but never --

Q Now, you have a background in race car engineering?

A Yes.

Q Can you tell me a little bit about that.

A Yes. I was a -- for a number of years what is called a Grand Prix Formula 1 mechanic at McClarin Racing in England, situated in Collinbrook, England. So I traveled around the world in the

Grand Prix circuit.

And then at the ripe old age of 29, I decided I had enough living out of a suitcase, basically. And that's back in '72, I believe. I started to pursue a career. I don't know how much you want to hear.

Q Well, can you tell me about your operational experience at nuclear reactors.

A I've never worked at a nuclear plant. My -- my nuclear background is with Westinghouse at first from out of college, and then -- I think for about six years, and then I was on the SDI project for a while on the so-called SB100 program, the space base reactor, and then the NRC.

Q Now, yesterday I understand that you accompanied others to the spent fuel pool at Millstone?

A That's correct.

Q Had you ever been there before?

A No.

Q Had you ever been in any other spent fuel pool before?

A No.

Q And can you please tell me your observations from that visit.

A Besides the obvious stringent security aspect of going through the various doors to get to the area, basically racks -- I saw racks there. I think it may have been new fuel racks, from where I was standing anyway.

Q How could you tell?

A I think I had somebody ask the question and I found out, I believe.

And I looked to see -- observed pipes for any means of dilutant, dilution, you know, in the area of pure water dilution aspect, where the fire pipes were and that kind of thing. I --

Q You said you looked for pipes or you saw pipes?

A I didn't look, I just noticed a lot of piping in the area. I wasn't sure what they were for. I heard -- I think it was Dr. Thomas (sic) talking about.

Q Dr. Thompson?

A -- Dr. Thompson, excuse me, talking about the heat removal system a little bit. But as I was about to listen to that, I was distracted by something else so I didn't get the answer that the gentleman he was talking to gave him.

Q But you did notice a lot of pipes?

A Yes.

Q Were some of these pipes overhead?

A Yes.

Q And do you know what was running through the pipes; was it pure water or was it something else?

A Well, the only -- besides the fire pipes, the fire extinguisher pipes I'm assuming, there was a big drain pipe from the roof of the building.

Q How could you tell it was a drain pipe from the roof?

A Oh, because I asked them and they told me it was.

Q Who did you ask?

A Mike -- he's sitting right back there.

Q Mike Jensen?

A Mike?

Q Jensen?

A I believe that's right.

Q And he told you it was a drain pipe from the roof that drained --

A From the -- drain water pipe. I think he called it a drain water pipe. Drainage pipe.

Q So you were distracted when you were asking about some of these pipes and you didn't pursue the questions?

A I was distracted when I was trying to listen. I wanted to listen to the answer that Dr. Thompson asked Mike, so I didn't get to hear the answer about the heat removal system.

Q And what did you learn about the heat removal system?

A Not very much. I didn't follow up anymore beyond that.

Q Did you happen to notice heating equipment suspended from the ceiling?

A No, I can't say I did.

Q Now, you mentioned that you saw racks for fresh fuel, I believe?

A Oh, I thought I -- at one end there were fresh fuel racks. I believe I heard right. I'm not 100 percent sure, but I think that what I heard was correct, that they were fresh fuel racks. Now, again, I didn't go up to the individual and ask him whether that was right or not.

Q You wouldn't have known just looking without the benefit of somebody guiding you what was a fresh rack and what was for spent --

A No, not really. No. No. No. This is the first time I've ever been, so I -- no.

Q What is your nationality?

A I was -- I was born in Malta, but I was raised in Australia.

MS. BURTON: Nothing further for this

witness.

MR. REPKA: Just a couple questions.

EXAMINATION BY MR. REPKA

Q Dr. Attard, Ms. Burton asked you about some pipes you may have seen yesterday in your tour of the spent fuel pool. Did you see any -- when you talked about some heating pipes, are you aware of those pipes? Do you know which pipes I'm talking about?

A No.

Q Okay. You said you saw pipes?

A I -- well, there are a lot of pipes, particularly along the wall.

Q Okay.

A And I was actually trying to look for pipes directly above the pool, that went off directly above the pool, where I presumed that if you had a leak in it, it would go straight down into the pool.

Q Right. And I think you mentioned you saw one pipe which was a drain pipe?

A Yeah.

Q The other pipes on the wall, how far were those from the spent fuel pool?

A Oh, well, they ran -- if my memory serves me right, there was one wall -- I only saw them -- except they came in from one building into the spent fuel pool and perhaps they went back out. I don't know. It looked like they came in through walls.

Q Were they right next to the pool or were they 20 yards, 30 yards?

A The one wall that I'm thinking of is where the transfer canal is. Again, I -- you're asking for me to check my photographic memory here which is not very good.

Q Was it a few inches away or a --

A Oh, the pipes were directly kind of nailed -- not nailed, but bracketed to the wall.

Q Right.

A So they were secured to the wall.

Q How far from the pool?

A Well, if they were -- in the transfer canal area, they -- I don't know, I would say maybe two -- two feet. I mean, they were against the wall. And the next area -- the next area to the -- where the transfer canal -- you know, whether it dripped or trickled down or whatever, it would eventually find its way, I would think, if they were water pipes. But I don't know what they were.

Q Did you have any particular concern that the pipes would leak to lead to dilution of the pool?

A No, not really.

Q Why not?

A First of all, Mike was telling us that -- that there are alarms in the control room, so if water level rose beyond a certain point, the alarm would go off, or if it drained, it would also alarm.

MR. REPKA: Okay. Okay. No further questions.

MS. HODGDON: I don't have any questions.

MS. BURTON: Okay. Thank you very much.

THE WITNESS: Thank you.

(Time noted 12:45 p.m.)

* * * * *

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of:	:	Docket No. 50-423-LA-3
	:	
Northeast Nuclear Energy Company	:	
	:	
	:	
(Millstone Nuclear Power	:	
Station, Unit No. 3	:	MAY 11, 2000

DEPOSITION OF DR. ANTHONY C. ATTARD

____

DR. ANTHONY C. ATTARD

Subscribed and sworn to before me this _____ day of ___________, 2000.

____

Notary Public

My Commission Expires:

EXHIBIT 27

Transcript, Deposition of Laurence Kopp, Ph.D., May 11, 2000

UNITED STATES OF AMERICA.

NUCLEAR REGULATORY COMMISSION

In the Matter of:	: Docket No. 50-423-LA-3
Northeast Nuclear Energy Company	:
Millstone Nuclear Power Station, Unit No. 3	: MAY 11, 2000

DEPOSITION OF LAURENCE T. KOPP, Ph.D.

CERTIFIED COPY

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For Northeast Nuclear Energy Company

NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555
BY: Ann P. Hodgdon, NRC Staff Counsel

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* * * * *

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L A U R E N C E   T .   K O P P ,   P h . D . ,

of United States Nuclear Regulatory Commission, Mail Stop 010B3, Washington, D.C., 20555, a witness in the above-entitled action, having been duly sworn by Kathryn Orofino, a Notary Public within and for the State of Connecticut, was examined and testified on his oath as follows:

* * * * *

MR. REPKA: Okay. These depositions today, the following stipulations will be in effect, consistent with the depositions that Northeast Nuclear took yesterday of their expert witnesses for the Coalition Against Millstone.

These depositions are to be used for discovery or as evidence in this proceeding only. Objections or motions to strike will not be considered to be waived except as to matters of form. The Deponents will be given a right to read and sign the transcript when it's complete, and the original of the transcript will be forwarded to the deposing attorney who will provide the opportunity for the witnesses to read and sign, and then the original will be filed with the Commission in accordance with the Commission's Rule of 10 CFR Part 2.

MS. BURTON: Okay. Good morning,

Dr. Kopp.

THE WITNESS: Good morning.

EXAMINATION BY MS. BURTON

Q Dr. Kopp, you have provided an affidavit in these proceedings that is dated April 10th of the year 2000, correct?

A Yes.

Q And you have had a role in assisting general counsel to the NRC with respect to discovery matters in this proceeding, correct?

A Yes.

Q Do you consider yourself an expert for those purposes?

A Yes.

Q Do you have a curriculum vitae?

A Pardon?

Q A resume, curriculum vitae.

MS. HODGDON: Is --

THE WITNESS: I thought it was attached to it.

MS. HODGDON: It is. I believe so. It is.

MS. BURTON: I'd like to ask for one. I don't have one attached with my documents.

MR. REPKA: We may have a copy here.

MS. BURTON: I have Dr. Attard.

DR. ATTARD: Yes.

MS. BURTON: I have his.

MS. HODGDON: I might have one, but I'm not sure.

MR. REPKA: I have one. I have a copy here if you would like to use it or make a copy. Do you need it to ask your questions?

MS. BURTON: I would like to see it. Thank you very much. Actually, when we take -- I'll make a copy of this. I'm afraid it will end up with my notes, so I'll give it back to you.

MS. HODGDON: I have a copy.

MS. BURTON: You do have a copy.

MS. HODGDON: Would you like to borrow mine?

MS. BURTON: It's your only copy?

MS. HODGDON: Yes.

MS. BURTON: I'll make a copy at the break.

Q Now, Dr. Kopp, according to your affidavit here, you have provided particular assistance with regard to certain interrogatories, correct?

A That's right, yes.

Q Beyond that, were you of other assistance in the preparation of the responses that we've received from the NRC?

A Yes.

Q And can you please elaborate how -- what your role has been in assisting with the discovery responses?

A Sure. Dr. Attard was the reviewer for the Millstone 3 pool expansion, and I reviewed it also, not in as much detail as Dr. Attard, and was also asked to provide responses to the three contentions that were filed, which I -- I have.

Q Well, let me then go back in time. You said that Dr. Attard reviewed the expansion application. Was that before there was an intervention petition?

A I'm not sure what the timing was before -- as far as when the application amendment came in and what the contentions were. I'm not sure of the dates.

Q Well, you said that he reviewed the application?

A That's right.

Q And then you took a further step beyond that with regard to the application prior to the intervention?

A I'm not sure if it was prior to or after the contentions.

Q All right. Well, let's confine ourselves to yourself because you will know about what you have done.

When did you first become involved with the matter of this pending application for reracking at Millstone Unit 3?

A I'd say three or four months ago.

Q And that would be approximately what month, what date?

A Either January or February of this year.

Q How were you assigned to this particular matter?

A I was asked to assist Dr. Attard, because I had previous experience in similar rerack projects and with the history of many of the regulations governing spent fuel pools.

Q Who asked you to assist Dr. Attard?

A I believe it was my -- probably my section leader at the time.

Q And who was that?

A At the time, it was Eric Weiss.

MS. BURTON: Let's just hold everything for just a moment. We have some arrivals.

(Recess taken)

BY MS. BURTON:

Q Dr. Kopp, what was the name of the individual who gave you that particular assignment?

A I believe it was Eric Weiss, W-e-i-s-s.

Q Now, can you tell us what you did to acquaint yourself with the issues here?

A Well, I reviewed the amendment request that came in, and I believe this -- this may have been Dr. Attard's first review of a request like this. And since I had done many of them in the past, I was asked to sort of overview his review and do a preliminary review myself, not as detailed as his, but to acquaint myself with the facts.

Q So you reviewed the application itself?

A Yes.

Q What else did you review?

A The -- I guess there was a prehearing conference, the contentions that were filed, and the technical specification changes that were requested.

Q Were you at all involved in the process in the evaluation of environmental assessment or the finding of no significant impact?

A No.

Q Or do you know if Dr. Attard was involved at that stage in this matter?

A I don't know.

Q So you were asked to be involved in the discovery part of this?

A With regards to the criticality concerns, contentions.

Q Would it be fair to say that your work here is in some way related to the staff's formulation as to its position on this license amendment application?

A Could you --

Q Do you see the two matters as being separate or all together? Do you understand that we are here today in a pending discovery proceeding?

A Uh-huh.

Q And that ultimately, the staff will arrive at a position as to this license amendment application. Are you providing input in that process as well?

A Well, I guess I will see Dr. Attard's final safety evaluation and be asked to comment on it if appropriate, so in that sense, yes.

Q Now, do you have the interrogatories with you that you provided an affidavit about?

MS. HODGDON: Yes.

A You're speaking about the April 10th document?

BY MS. BURTON:

Q Correct.

A NRC staff response?

Q Right.

A Yes.

Q And you have identified that you provided assistance to a number of these interrogatories. Particularly addressing yourself to Interrogatory F-1, do you have that before you?

A Yes.

Q Now, can you tell us, please, what you did in order to determine that -- the information base that is requested here?

A I don't recall reviewing all instances of various errors of movement in managing moving and tracking spent or fresh fuel at Millstone.

Q You have not done that?

A No. I am aware of reported licensing event reports, but not primarily at Millstone, but at many plants.

Q Well, let's confine ourselves for a moment to Millstone. Which ones are you familiar with at Millstone?

A I don't know of any errors at Millstone that involved misplacing spent fuel involving any criticality concerns.

Q Are you aware of any LER's that concern spent fuel pool incidents at Millstone?

A I recall there was an incident involving adequate cooling, but I was not involved in that time because it was not in my area of expertise as far as criticality.

Q And which event was that, do you recall; which plant, approximately what time frame?

A Well, I believe this was about a year and a half ago or two years ago.

Q How did you become aware of that if that was not within your area?

A It was just in public press releases.

Q And are you aware and familiar with any other LER's concerning the spent fuel pool at Millstone?

A As far as?

MS. HODGDON: Excuse me. Do you mean all spent fuel pools at Millstone?

MS. BURTON: Yes.

A As far as criticality concerns, as far as --

BY MS. BURTON:

Q As far as any License Event Reports that were filed with the NRC.

A I recall several years ago a -- an LER that was filed concerning Boraflex degradation at one of the Millstone plants and seismic concerns regarding Boraflex degradation.

Q And tell me what you recall about that. Are those two separate incidents, or was that one?

A One. One incident.

Q Uh-huh. Well, can you tell us about that one; when did it occur, which plant, and so on?

A I'm not sure which unit it was. It was probably either 2 or 3, because it was a PWR concern. And the -- what exactly did you want to know?

Q Well, I wanted you to tell us what familiarity you have with License Event Reports filed concerning the spent fuel pools at the Millstone station.

A I recall that there was an event filed that had to do with concerns about a seismic event embrittling or detaching the embrittled Boraflex material that was attached to the spent fuel racks.

Q Do you recall when that occurred?

A I'd say sometime within the last five years, but probably closer to five than one or two.

Q And what was your involvement with that particular LER?

A I reviewed it, and I guess we asked some questions about licensing, and I think we issued a -- probably an NRC Information Notice on it.

Q Did you review that event in conjunction with your assistance in the interrogatories in this matter?

A I have seen reference to it, came across a reference to it as part of a discovery.

Q You said that occurred, to the best of your recollection, within the past five years. Can you be a little bit more precise?

A I'd say probably just about five years.

Q So those are two LER's. Have there been others?

A For Millstone, at Millstone?

Q Millstone.

A I believe there is one maybe longer than that, about seven or eight years ago, that had to do with some calculational discrepancies in the spent fuel pool.

Q Which pool; do you know?

A I don't recall the unit. I don't recall which unit.

Q And what role did you play in that license event matter?

A I believe there was a prehearing conference that I attended, and I think the matter was -- was cleared up at the -- at the prehearing.

Q What do you mean by "cleared up"?

A Well, it didn't go any further than -- than -- than the prehearing conference.

Q Was there a determination --

A There was a board there, the NRC three-man board, judges.

Q Was there a determination that there had been a discrepancy?

A There was a determination, I believe, that there had been a discrepancy, and that the calculations were performed with the better model, and there was no -- I guess the final outcome was there was no safety significance that was attached to it.

Q So that's three LER's. Have there been others?

A I don't recall any others, no.

Q Now, you have a lengthy history of service with the NRC. It's about 35 years or so?

A Yes. Yes.

Q And do you consider yourself an expert in the area of criticality at spent fuel pools?

A Well, I have worked on them for the last 20 years, so I -- I would say, I guess, yes.

Q And, in fact, you prepared a memorandum dated

August 19th, 1998, on the issue of criticality analysis at spent fuel pools?

A Yes.

Q And, in fact, that has been produced in these proceedings by the applicant. Are you aware of that?

A That is the memo from myself to Timothy Collins? Yes.

Q Can you tell us why you prepared this memo?

A Yes. The previous guidance for spent fuel pool analysis, I guess, was about 20 years old. It was ~~in a so-called Grimes letter that was sent to all~~ licensees. And several things had progressed from then that the Commission had accepted which was not in the Grimes letter, and that was just an update of the current practices that were acceptable to the NRC.

Q With regard to fuel handling and the spent fuel pools?

A With regard to criticality and spent fuel pools, yes.

Q Now, you say that this was an update. When was the issue of criticality and spent fuel pools first addressed by the NRC or the AAC, if you know?

A I guess from the date of the first nuclear plant.

Q And that would have been when?

A I would say somewhere in the '60's.

Q So as of 1998, there was perceived to be a need to update the staff -- I guess this was directed to the staff -- with regard to criticality issues at spent fuel pools, correct?

A Yes. This served two purposes; to update the current acceptable methods that the staff had reviewed and approved over the years for spent fuel pool criticality, and also as a guidance for new staff members who may have recently come in who weren't aware of all the progressively accepted techniques.

Q So over a period of decades, acceptable methods -- the term "acceptable methods" has been developing; would that be fair to say?

A Yes.

Q There's been an evolution?

A Yes.

Q Can you describe the evolution in terms of safety standards?

A Sure. In the early '80's is when licensees started taking credit for so-called burn up, burn up credit for spent fuel production and reactivity caused by burn up.

Around the same time some licensees went to sort of a checkerboarding arrangement of fuel storage configurations so as to increase the spacing, because at the same time uranium enrichments were increasing in the fuel assemblies. Reactors are going to longer cycles and fuel enrichments are increasing, plus there was no central repository and more and more fuel assemblies had to be stored on site, so there had to be some techniques to be able to manage it safely.

Q Now, has the K-effective standard changed over the years?

A That has always been .95 as the design basis, as far as I know.

Q Do you know if .9 was ever used and practiced?

A I have seen .90 in some older boiling water submittals and tech specs.

Q So the change from .9 to .95, how would you characterize that in terms of strictness; more strict or less strict?

A Well, it goes from 10 percent margin to criticality to 5 percent margin to criticality, so it still offers a significant margin, it just reduces the margin.

Q Was there a time when open frame racks used to be required?

A I'm not sure what you mean by open frame racks. They are all open. They all have openings in the top where the fuel assemblies go in.

Q Was there a time when there was a more particular requirement for a certain type of rack to be used?

A Well, are you talking about based on spacing alone or --

Q No, not based on spacing alone, just generally.

A Well, the racks have evolved, as I said, with the evolution of higher enriched fuel and the increase in the number of fuel assemblies on site. Racks have had to be compacted so that more fuel assemblies could be stored.

Q And what has that done -- how would you characterize what that has done in terms of the margin of safety?

A Pardon me, concerning what?

Q The margin of safety. Has it increased or decreased the margin of safety?

A It has kept the margin of safety the same. They are still designed to K-effective of .95.

Q Now, in the interrogatories, there is a question concerning an analysis of the probabilities and consequences of a criticality accident in spent fuel pools. And the answer that is provided here is that there has not been such a study by the NRC.

A Yes, there has not been.

Q You're aware -- I think that was one of the answers that you assisted with. Can you tell us when there will be such a study by the NRC?

A I don't think there will be. The NRC has a practice of including criticality in the design, and therefore, we don't feel there is a need to design assuming that there is a criticality event.

Q Well, but there -- has there been an analysis to arrive at that conclusion?

A Well, all the plant analyses arrive at that conclusion by maintaining a safety margin of at least 5 percent subcriticality even for the worst accident.

Q Let me go back to where we were a moment ago on the LER's. The information that you have with regard to Millstone is based on LER's, as you have said.

Do you have any other information about incidents at the spent fuel pools at Millstone in terms of boron dilution or fuel mishandling?

A No, I don't. I know of no boron dilution events that occurred at Millstone that result in an LER. I'm sure I would have been aware of that if there were.

Q Can you tell us what the threshold standard is for filing an LER on the part of a utility in the event of a boron dilution or maybe an event of a boron dilution?

A Well, I'm not sure what the regulations are for filing LER's. I would imagine it would involve a technical specification violation, first of all, or a violation -- or a decrease in margin in the licensing basis analysis, something like that.

Q But you're not sure.

A I'm not sure what the exact requirements are for filing an LER.

Q Do you know if there's a requirement for filing information about boron dilution that does not violate tech specs?

A I'm not certain. I don't know.

Q Do you know what the reporting requirements are with regard to fuel mishandling?

A There's fuel -- fuel storage configurations in spent fuel pools are the only part of tech specs, so any violation or fuel mishandling accident or fuel misplacement would be reported, because it would be a tech spec violation.

Q Sorry. Could you go through that one more time.

A Technical specification is usually defined the storage configuration of fuel in the spent fuel pool. If there is a misloading which would violate those configurations, that would be a technical specification violation, or at least a violation of the design basis analysis, and that would require some type of report.

Q So you would assume that in all cases after fuel mishandling, that they would be required to be ~~reported to the NRC by the utility?~~

A I'm not certain. If it involved a technical speculation violation or a violation of licensing design basis, I would assume so.

Q But you don't know for sure.

A No.

Q Addressing your memo of August 19th, 1998, again, can you tell us what database you used to -- for the information that you relied on for the preparation of this memo in terms of incidents at spent fuel pools?

A In terms of incidents?

Q Uh-huh.

A I -- it was not based on incidents at spent fuel pools, it was based on regulatory requirements over the years, staff guidance over the years, methodologies that had been approved over the years.

Q Well, does the NRC maintain a database of incidents of boron dilution or fuel mishandling at the spent fuel pools at Millstone and elsewhere?

A No, we don't.

Q You do --

A These are available in independent or individual LER's, but as far as maintaining a database, I don't know that we do. I don't think we do.

Q If you did, do you suppose you would be in a position to be aware of that, given your 35 years of service and your specialty in this area?

A I'm sure I would.

Q So would you state with confidence that there is no database that the NRC has compiled with regard to boron dilution and fuel mishandling incidents at commercial nuclear reactors in this country?

A I don't know of any, so I'm sure if there were, I would have known of it.

Q And in the same way, are you sure that you would have had access to information about any boron dilution or fuel mishandling events at Millstone prior to compiling your memo of August 19th, 1998?

A Yes, any incidents that would involve safety or have any safety significance, I would have.

Q And who would have made the determination as to whether incidents involved safety significance?

A Could have been a number of people, including myself.

Q Well, would it be the NRC that would determine that, or would it be the utility?

A Well, the utility may make a determination of whether there was safety significance or not, but the NRC would do an independent review and make an independent determination.

Q Well, let's say the utility makes a determination that there's not a safety issue and therefore, doesn't report to the NRC, how does the NRC then find out about the incident?

A Well, we have resident inspectors at all the facilities, at all the sites. We have regions that oversee various plants and their locales, so it would be one of those means.

Q One of those means what?

A Of knowing whether -- of being able to detect something like that.

Q So you're assuming that in all cases where there have been incidents of fueling mishandling or boron dilution at reactors that the resident inspectors would have evaluated those for safety analysis and reported them to the staff in such a way that you would have had access to that information in preparing your August 19th, 1998, memo?

A No, I'm not sure of that, but one of the means of being able to -- to know whether such an event happened would be through a resident inspector. Not necessarily the NRC staff at headquarters.

Q Can you tell us why the NRC does not maintain a database of boron dilution and fuel mishandling at the reactors?

A Because those are two events that are analyzed in the design basis, whether they can occur or not. And for a complete boron dilution, I can't really think of any mechanism that would cause that, but the licensees are required to analyze for that anyway, and show that there's still 5 percent criticality margin.

Q You're a scientist, are you not?

A Yes.

Q In order to scientifically analyze these issues, you need to have a proper database as the industry evolves, wouldn't you think?

A Yes, and I am only aware of one reported incident of boron dilution event, which was very minor.

Q And where was that?

A I am not sure which plant it was, but I think it was brought up by either Dr. Thompson or Mr. Lochbaum yesterday and at the prehearing.

Q Was that at McGuire?

A It might have been McGuire.

Q Was that the reference? You were not previously acquainted with that event?

A Yes, I was aware of it, but it didn't amount to very much. You consider the spent fuel pool is about 30 percent subcritical and you dilute about 100 or 150 ppm of boron, that only brings K-effective from .7 to .71. That's not very significant safety-wise. Going from 30 to 29 percent subcritical, I can't get very excited about that as a safety concern.

Q Do you know if the NRC maintains a systematic record of administrative failures at the reactors?

A No, I don't. That could encompass a very large area of --

Q If there were such a systematic record, would you be aware of it?

A I'm not sure if I would be. I'm not sure if my area of the organization would be aware of it or would maintain that.

Q But you're not yourself aware of such an --

A I'm not aware of it.

Q Are you familiar with an industry wide effort to establish a database of events concerning boron dilution and fuel mishandling?

A No, I'm not.

Q You're not?

A I never heard of it, no. I'm not saying that it might not be a good idea, but I have never heard of it industry wide.

Q Dr. Kopp, are you aware that the NRC requires licensees to perform analyses of the probability of degraded core reactor accidents?

A I'm sorry, can you repeat that again.

Q That the NRC requires licensees to perform analyses of the probabilities of degraded core reactor accidents?

A I'm not aware of that, no.

Q Well, are you aware that planning for off-site emergency response assumes that a degraded core reactor accident could occur?

A No, that's -- I'm not involved in that. That's not my area. I'm aware of --

Q Well, are you aware of it?

A No. I'm aware of safety analyses for various events that are required as part of Chapter 125 of FSAR's, which go anywhere from anticipated operational occurrences all the way through to severe accidents. I'm not aware of any -- I'm not involved, at least, in any of the probability assessments of various core accidents.

Q The question is are you aware?

A No.

Q Are you aware that K-Efficient says the NRC will not conduct an analysis of the probability of consequences of the spent fuel criticality because the policy is to prevent criticality?

A Yes, I believe that's what I just said earlier.

Q Then, can you tell us why --

MR. REPKA: Excuse me, where are you reading from?

MS. BURTON: Where am I reading from?

MR. REPKA: Where is that --

MS. BURTON: I'm looking at some notes.

MR. REPKA: Okay. You weren't looking at a document.

BY MS. BURTON:

Q Can you tell us, why is criticality not addressed in the same way as are degraded core reactor accidents?

A You mean in spent fuel pools?

Q Yes.

A I would think, first of all, cores are designed for criticality. I mean, that's how you generate power. You have to have a reactor going critical. And if you have an accident condition during a power generation or something like that, you have a much more severe event than trying to increase reactivity on somewhat of a -- in a spent fuel pool, which is quite a bit subcritical. It's just two different animals.

Q And that is your explanation of why criticality is not addressed in the same way as are degraded core reactor accidents?

A Yes, they are just two different -- two different animals. A core, a reactor and a spent fuel pool, one is designed for criticality, and one is designed to be inherently subcritical.

Q Well, one has been analyzed and the other has not, isn't that fair, at least by the NRC?

A Analyzing criticality in a spent fuel pool is a very difficult analyses. I'm not sure it could be done even with current techniques.

Q Would you explain why.

A Just assuming an initiating mechanism would be difficult. One cannot think of any initiating mechanisms that could cause criticality in a spent fuel pool, at least I can't.

The biggest reactivity condition to a spent fuel pool would be a boron dilution event. The boron holds out about 25 percent of the reactivity in a spent fuel pool, and if you lose all the boron, calculations still have to show that you're still 5 percent subcritical, plus it's a very slow event, slow reactivity addition event.

If you consider feedback effects, probably the final state would, even if you did -- were able to go critical, would be just a chugging along or the boiling of some water. Not a very major event.

Q Dr. Kopp, have you read Appendix C of Orange County's filing of January of the year 2000 in the pending matter involving the Shearon Harris plant?

A I read the submittals. And I can't quite recall what Appendix C referred to, but I'm sure I read it all, yes.

Q And is your statement informed with the information that you gleaned from that appendix?

A I don't know. If I can see the appendix. I'm not sure what was in it.

Q This is Appendix C, which is entitled, "Assessing the Probability and Consequences of

Criticality Events in Fuel Pools," which I'll show you.

A I recall seeing this several months ago during the Shearon Harris.

Q So you have had occasion to review this appendix?

A Yes.

Q Do you agree -- can you tell us if you agree with the conclusions of this analysis, or disagree?

A Is there a section on conclusions?

MS. HODGDON: It might be more helpful if you were to ask him which conclusions you're talking about.

A Yes, I do not agree with these conclusions.

BY MS. BURTON:

Q Can you break it down and be specific as to why not?

A Well, the first paragraph implies that a one-time variety of administrative controls are acceptable, and then it talks about fuel misplacements. You select fuel to be placed in a given position in a spent fuel pool on a one-time basis, so that would seem to fit in with this allowance of one-time administrative controls.

And it goes on to speak about multiple events, which is not what the double contingency principal that was brought up yesterday refers to. The double contingency says that -- realizes that there can be two events or more that can cause criticality, and therefore, the spent fuel pool is designed so that the worse unlikely event would not cause criticality, and they have a sufficient margin.

In other words, combinations of various events like fuel misloadings and boron dilutions are not the way the double contingency principal is intended.

Q So you have a difference of opinion there.

Are there other differences that you have with this appendix?

MS. HODGDON: Objection. Dr. Kopp, it appears, is being asked to agree with everything that he doesn't specifically disagree with, and I don't believe that it's appropriate to put him in that position. If you want to ask questions, specific questions regarding this, I would suggest that he be given an opportunity to read it and then ask specific questions with regard to specific conclusions.

In other words, I would not -- I would suggest that his failure to disagree is not a wholesale endorsement of this document. I would like that on the record.

MS. BURTON: I think it was clear from my question and his response.

MS. HODGDON: I don't believe it was.

A I can go on. The next paragraph talks about the experience at U.S. nuclear plant showing the fuel mispositioning involving misplacement of a fuel in one or more inappropriate burnups is a likely occurrence. We don't necessarily agree that this is a likely occurrence, but that is an event that's analyzed anyway, to show that pools still maintain the 5 percent subcriticality margin.

The next paragraph talks about experience showing that the concentration of a soluble boron in a pool can fall below specified levels. I don't know of any events except the one that was previously mentioned where the concentration fell to about 150 ppm, which I said ranged it from a subcriticality margin of about 30 to about 28 or 29 and a half percent, which is still not significant.

Plus the fact that a complete boron dilution event is part of the design basis of each plant. As I said, they analyze the spent fuel pool configuration with pure water rather than borated water, and that will still show that there's still at least a 5 percent subcriticality margin.

BY MS. BURTON:

Q Go on. I've asked you to break it down into the specific issues that you take issue with. Are there others?

A I think that's every paragraph that's been in there. There's a mention about calculations performed show that supercritical configuration could occur if two or more fuel assemblies are positioned and the concentration of soluble boron is reduced. Those are the two or more accidents that I spoke of earlier that are not required by double contingency.

Fuel mispositioning or dilution of soluble boron will occur as a result of failure of ongoing administrative controls. And there's a mention that there have been several experiences shown that there are -- have been fuel mispositioning events.

Well, I think that illustrates more than anything that administrative controls work, or we wouldn't have known about these fuel mispositioning investments. These fuel mispositionings were found detected and corrected because of administrative controls.

Q Were they always found and detected immediately?

A I'm not sure if -- what do you mean by "immediately"? As soon as they occurred or before there can be any type of criticality?

Q As soon as they occurred.

A No, I don't think so. I don't believe so.

Q So in some cases, you're aware that there have been lags in the failure of discovery of administrative control?

A I'm sure there have been failure in discovery that a fuel assembly was mispositioned, yes.

Q So that would be another failure of administrative control; wouldn't it?

A Well, as I said, the fact that they are eventually found shows that eventually the administrative controls did work.

Q And that's a good enough standard, you think, for spent fuel pools?

A Well, it is when you consider the fact that mispositioning of fuel assemblies are determined and calculated to be -- to show that there's no safety significance if they occur in a spent fuel pool. This is one of the events that are required to be analyzed by the staff.

Q Do you dispute that supercriticality could occur, given the assumptions stated in Appendix C?

A In the combination of fuel misplacements and boron dilutions?

Q Right.

A There could be various combinations, sure, that could result in criticality well beyond anything realistic, and they are well beyond what's required to be analyzed. And I'm not sure what one would do with information like that when developed in an envelope like that. If it took 100 misplacements and the 75 percent boron dilution event to show that you can go critical, I'm not sure what that would show.

~~As I said, we look at the worst most~~ unlikely event, complete boron dilution event, and still show that there is a 5 percent subcriticality margin.

Q Can you tell us if a degraded core reactor accident was considered realistic in 1970?

A I don't know.

Q You do not know?

A I'm -- when you speak about the degraded core reactor accident, I'm not sure what you're referring to. I mean, we've looked at the reactor core accidents since the beginning of nuclear power plants.

Q In the year 1970, were they considered realistic by the NRC?

A They were classified according to whether they are expected to occur, say, during the lifetime of a plant, or whether they were not expected to occur, but were analyzed anyway, such as the accident, such as rod ejection accidents, loss of coolant accidents, compared to the other end, some minor transients, such as rod misalignment, control rod misalignment. So these were looked at, as far as I know, since nuclear plants were designed and built.

Q Are you familiar with NRC Regulatory Guide 1.174?

A I'm not sure. What is the title?

Q Concerning risk assessments.

A No.

Q Are you familiar with individual plant examinations and individual plant safety assessments?

A No.

Q Are you familiar with INPO?

A Is that an organization?

Q Yes, uh-huh.

A I've heard of them.

Q Are you familiar with data that is collected by INPO concerning events in spent fuel pools which may not reported to the NRC?

A No, I'm not.

Q Are you familiar with the term "standard review plan"?

A Yes.

Q What is that, please?

A Well, that was a set of guidance that the staff developed for licensees or applicants that were coming in for a construction permit or operating license for a nuclear power plant. And it was guidance on how to calculate various accidents, the assumptions to make, the results that were acceptable to the staff.

Q And when was the first standard review plan implemented; do you know?

A I'd say the early '70's.

Q And what analysis of boron dilution and fuel mishandling in spent fuel pools are required to be submitted in conjunction with those review plans?

A Well, as I said, the standard review plan for the spent fuel pool requires that the analysis be done in pure water, so the requirement is that the analysis be done assuming complete boron dilution without describing what mechanism may be available to -- to allow that to happen.

Q Is that the extent of the analysis that is required of boron dilution and fuel mishandlings in the standard review plan?

A In the spent fuel pool?

Q Yes.

A There are boron dilution events in the core that are reviewed.

Q What about fuel mishandlings?

A I'm not sure if the standard review plan mentions anything about fuel mishandlings or misplacements. There is, of course, the fuel assembly drop event, which is not really a criticality concern. That's more of a radiation concern. The concern is you drop a fuel assembly and you damage other fuel assemblies in the pool and damage the cladding and release nuclides, so that's more of a dose concern rather than a criticality concern.

Q Dr. Kopp, what is your plant operational experience?

A Plant operation?

Q Operational experience.

A I have never worked at a nuclear plant.

Q Now, do I understand that yesterday you participated in a site visit of the Unit 3 spent fuel pool?

A Yes.

Q Had you ever been there before?

A Not at Millstone, no.

Q Can you tell us your observations, please.

A All of them? I'm -- I'm not sure what you mean. What are you --

Q I'd like to hear your impressions of what you observed at the spent fuel pool.

A Well, I observed that security was very stringent, I observed the space in the spent fuel pool where the new racks will be going.

Q And what did you observe about that?

A First of all, that there was adequate space to place the spent fuel pool racks, and that the new racks could come in and be placed in the vacant position in the pool without going over the existing spent fuel, so that there would not be concern about a rack drop accident on spent fuel that is currently in the pool.

Aside from that, there's not very much to get excited about looking at a spent fuel pool. There's not really much going on, so I'm not sure what you're after as far as my observations go.

Q Did you notice anything about the location of the area for fresh fuel in relation to spent fuel?

A When it first comes in or when it's put into the pool?

Q The location in the pool.

A Yes.

Q You did make an observation as to the location in the pool for fresh fuel in relation to spent fuel?

A Yes, I believe it was in an area of the pool that was checkerboarded. It was either capped, or some way to prevent fresh fuel from being placed in every -- every pool location. It was checkerboarded in order to increase the spacing between fresh fuel.

Q Was there anything about the spacing that particularly was of particular interest to you or concern?

A No, except that the spacing between fresh fuel was about twice as -- twice the distance as it was between the spent fuel.

Q In terms of the location of the fresh fuel with reference to the spent fuel, did you make any observations about that or have any concerns about that?

A Well, of course, it was closer to the reactor because it eventually has to go into the reactor core, so it was near the reactor opening. Aside from that, there were really no other observations.

Q Did you happen to notice the overhead heating system?

A No, I -- I've seen -- I saw pipes and all types of things, equipment in there, but I'm not sure what was what. My expertise is in criticality and the pool, not in the auxiliary systems, so I wouldn't have any comment on any of that.

Q But you did observe overhead piping?

A I observed piping, yes, overhead.

Q But you don't know --

A What was going through there.

Q -- what was going through the pipes?

I wonder if you can tell us what NRC guidelines there are and what criteria there are for determining when administrative controls are appropriate and when not?

A Are you talking with regard to criticality or --

Q With regard to the spent fuel pools.

A The NRC has a lot of administrative controls for -- since I've been involved in spent fuel pool reviews for placing fuel assemblies.

Q Would you please identify the guidelines and criteria that are used by the NRC, if you can, if you know?

A I'm not just sure what you mean by guidelines and criteria.

Q Well, are there any standards that the NRC applies when it evaluates whether reliance on administrative controls is appropriate?

A Well, we have regulations that allow administrative controls for precluding criticality.

Q You're talking what appears in the CFR?

A Yes.

Q Apart from that, standards, guidelines, criteria?

A There are various standards that -- ANSI standards. I'm not sure of the number or the exact titles, and I'm not even sure if they have been officially endorsed by the NRC, but there are industry standards that -- that speak about administrative controls.

Q Okay. But my question was NRC standards, not industry standards.

A Our NRC standard is there would be -- one of them would be the regulation 10 CFR 50.68.

Q Well, apart from that, are there any other standards that you can identify? And are there standards that the NRC itself employs; if you know?

A Offhand, I don't know, no.

Q Well, if there were standards, you probably would have referred to them in your memo of August 19th, 1998; is that a fair statement?

A Possibly there would have been a reference in that memo.

Q Is there a reference in your memo?

A To a standard?

Q Uh-huh.

A Not that I know of.

Q So how does the NRC, then, decide where to draw the line where it has to decide whether an administrative control is appropriate or not in a given situation involving spent fuel pools; if you know?

A I'm not sure what you mean, where the NRC draws the line.

Q You don't know then?

A I'm not quite sure what you're -- what you're asking.

Q Well, in any given case where there is an application to rely on administrative controls and the NRC has to decide whether it's appropriate or not in that instance, I'm looking to understand what the NRC uses in order to understand whether it should allow administrative controls or say no, they would not be appropriate in a certain case?

A Well, for one thing, we would probably look at how many things would have to go wrong for the administrative controls in order to get to a situation where, for example, an erroneous fuel assembly could be put in the wrong place, whether it would take several operators not following a set of preplanned procedures and a second set not verifying the final position of a fuel assembly. So things like that would be looked at. How many screwups there would have to be to get into an abnormal situation.

Q So there is a screwup policy that the NRC employs? Can you identify a little bit better for us?

A Not any better than I just said. We would look at how many times -- how many controls, how many administrative controls there are on selecting the fuel assembly to be put in a certain spent fuel pool location.

Q Would it be fair to say that the analysis done is -- that the analysis that is done is done on an ad hoc case-by-case basis, without standards and criteria?

A There are no concrete standards or criteria, it's just a matter of the reviewer looking and seeing what would have to go wrong. And aside from that, the event is analyzed as a required analysis anyway. It has to be analyzed.

So the analysis for a spent fuel pool accident is somewhat different than a reactor core, because in a reactor core, you have to assume what the initiating events are, how long it takes, what controls there are.

Spent fuel pool, the requirement is that you assume the accident occurs independent of whether it can, how likely it is, how long it takes to occur, and so forth. It's instantaneously all the boron in the spent fuel pool is lost. How that's magically done, I don't know.

Q Are you familiar with an incident at Millstone where there was leakage that went undetected for a certain period of time leading to a drop in the pool level, the water level?

A No, I'm not.

Q Well, if that had occurred, would that be something that you would have considered as part of your work on this matter?

A That would be more of a radiation problem than a criticality problem.

Q Would it have been of interest to you and concern in your assessment and in your participation in this proceeding?

A It would, yes.

Q And can you tell us why; why that would be relevant to your role here?

A Well, depending on what the makeup source was for the water, if there were a pool level drop -- first of all, a pool level drop within a very small band is enunciated in the control room. That's one thing we heard yesterday at the site visit. So the operators in the control room would be aware of any pool level change through measurement systems and alarms.

Q Do you know when that system was put in place?

A No, I don't. All I know is that most spent fuel pools have both a low level and a high level alarm, which would indicate a several inch variation between what the required 23 feet of water above the spent fuel is -- whether it's decreased or increased.

Q Is there a criticality alarm at Millstone?

A I think it just went off.

In the spent fuel pool?

Q Yeah, at Millstone.

A I'm not sure there's a criticality alarm. There are radiation alarms in the spent fuel, which indicate a pool level drop or damaged fuel or spent fuel assembly that may be coming too close to the top of the pool, but those, again, are radiation concerns, not criticality concerns.

Q Do you know if there is any criticality monitoring that goes on at Millstone?

A I do not know.

Q You do not know?

A No.

Q Would you be concerned to learn that there have been incidents of fuel mishandling at Millstone that were not reported through the LER process?

A I would be interested. As far as concerned, I would have to know what the events were and how significant they were.

Q Would your participation in these proceedings potentially be affected by learning about a series of fuel mishandlings at Millstone?

A A series all at the same time or --

Q Over time.

A Over the years? Over time.

Q Over time.

A As I said, I would be interested in learning about it, but as I said before, this is an event that's analyzed for anyway, so --

Q But you have not analyzed -- your analysis has not been informed by fuel mishandling events at Millstone, other than what you spoke of earlier in the three LER's; is that correct?

A That's correct, as far as I can recall, yes.

Q Do you know if this amendment is granted, would it be possible for spent fuel to be moved from Unit 1 to Unit 3?

A I don't think so. I think I asked yesterday whether the units were connected, whether the spent fuel pools were connected, and I was told no. Whether it could be done via casks, dry casks, that's another question. But as far as being transferred under water, I don't think so. I don't think there's a connection.

Q I didn't say under water. I meant transferred at all.

A Oh, I assume that fuel could be put into a dry cask and transported over into another unit, spent fuel pool.

Q So you would assume that it would be possible to move Unit 1's spent fuel to the Unit 3 pool if this amendment allowing reracking were to be granted?

A Well, that would require separate approval, first of all, for dry cask storage and for dry cask movement. That would not be part of this amendment request. It's a separate type of request. It would be a different organization within the NRC that would be involved in that also.

Q If the spent fuel pool in Unit 3 were presently beyond capacity to allow a full core off-load, is that a situation that would require shutdown?

A No.

Q Can you tell us why?

A There is no requirement to have a full core off-load capability in spent fuel pools. I mean, that's up to the licensees if something occurs where they have to shut down the reactor and have to off-load the fuel, it's their concern whether they have -- whether they can do that or not. If they can't do that, the fuel would have to remain in the core, and it's completely a licensing decision -- I mean a licensee's decision.

Q Aren't there occasions, emergency conditions that could arise that would require full core off-load?

A Not that I know of. I don't see any reason where there were an emergency the fuel could not remain in the core. And as I said, that would be, you know, the licensee's problem whether he would -- if he couldn't off-load fuel that were damaged in some type of event in the core and had to keep it in the core, that would prevent him from operating again with -- with fresh fuel. But there are no regulations that require a full core off-load, as far as I know.

Q You mean capacity for a full core off-load?

A Right.

Q But would you agree that there are occasions when it is necessary to unload the reactor, defuel the reactor, other than a refueling of it?

A I can't think of any. Like I said, this would be purely a licensee's decision whether he would want to be able to off-load the core if some event occurred. If there is not capacity available for it, then licensees would be forced to maintain the fuel in the core and discontinue operating.

Q Are you familiar with the standards -- qualification standards for operators at the reactors?

A No.

MS. HODGDON: While you're shifting gears, could I ask, first of all, could we take a break; and secondly when you're contemplating lunch. How much more do you have?

MS. BURTON: I think I have just a little bit more with Dr. Kopp, if you would like to take a break now.

MS. HODGDON: I think a break for everybody.

MS. BURTON: All right.

MS. HODGDON: Thank you.

(Recess taken)

MS. BURTON: We're back on.

Q Dr. Kopp, you have indicated that you have reviewed the submissions by Northeast Utilities in this matter?

A Yes.

Q And is it your understanding that the licensee is proposing to maintain a soluble boron in the spent fuel pool?

A Yes.

Q And do you agree that it is needed in this matter; that it's necessary for the licensee to maintain soluble boron in the spent fuel pool at Unit 3; and if so, why?

A The main requirement to maintain it is -- primarily has to do with the core itself during refueling. When everything is connected -- the transfer canal, the spent fuel pool and the reactor core -- during refueling, when the fuel is removed from the core and put in the pool, and new fresh fuel from the pool is put in the core, if there are no boron in the -- there's a requirement to maintain at least a 5 percent shutdown margin in the core during that time.

And you need about 2600 ppm of boron to meet that. If there are no boron in the pool when everything was connected, that would be -- dilute the boron in the reactor core, which would not maintain the 5 percent required shutdown margin during refueling.

So that's one reason why you have to maintain boron in the spent fuel pool.

Q Apart from that, do you perceive any other reason why the soluble boron needs to be maintained in the spent fuel pool at other times?

A Do you mean any level at all or the required 2600 ppm?

Q Any level at all.

A As I said, the analysis is done assuming that there is no boron in the spent fuel pool, and that has to show that there is still a 5 percent subcriticality margin without boron.

Q Do you agree that boron is required to be maintained because of the potential for fuel misplacement?

A Well, that's one of the lesser events that I looked at. The complete dilution of 25 or 2600 ppm of boron from the spent fuel pool is by far a larger reactivity addition to the spent fuel pool than any fuel misloading event. The reactivity insertion due to fuel misloading event is on the order of a few percent reactivity, whereas boron dilution event is about 25 or 30 percent of reactivity.

So by far, a boron dilution event is the most reactive accident or the accident which has the most reactivity to a spent fuel pool. That is the limiting event.

Q Do you agree that it is necessary to maintain soluble boron because of the potential for fuel misplacement at Unit 3?

A I think the licensee or whole text analyses show that fuel misplacement would still not result in criticality, even with complete loss of boron.

Q Did that study analyze multiple misplacement incidents?

A I don't think so.

Q Are you aware that there have been multiple misplacement incidents at reactors?

A I recall seeing something where there have been several, yes.

Q Do you agree it would be important in this matter for an analysis to be undertaken that would postulate multiple misplacement incidents in the spent fuel pool?

A That would go beyond what the staff requires in reviewing -- in spent fuel pool analysis.

Q And could you point to the standard or the policy that provides that?

A There's one thing, the letter -- the letter you referenced before, my letter to -- to Mr. Collins, which addressed all the approved methodology and guidance for spent fuel pool criticality analysis, which talks about the single fuel misplacement.

Q Well, where is the standard that establishes that that is the extent that will be required by the NRC?

A There is no standard. The -- the guidance is provided by the double contingency principal.

Q Do you agree that you drafted your memo of August 19th, 1998 in part because there was confusion with respect to this issue on the part of licensees and the staff -- on the part of the staff, shall we say?

A On the part of the staff, no.

Q Then what was the need to draft the memo?

A As I said, there were new methodologies that had been reviewed and approved from the previous guidance that had been issued, which was the so-called Grimes letter.

More recently, we approved a methodology for accrediting boron, partial boron, in spent fuel pools, which Millstone doesn't use, but other licensees have taken credit for partial boron to meet the .95 criterion.

Q But again, your memo does not -- stricken.
So you're saying that because the staff has never required an analysis of multiple misplacement incidents, does that alone set a standard?
A No, I'm saying the double contingency principal, which has been adopted by the staff for many years, does not require it.
Q And you agree that analysis has not been done with respect to this present application?
A For multiple misplacements?
Q Yes.
A Correct, yes.
Q Now, going back for a moment again to the site visit yesterday of the spent fuel pool, when you were there, were you able to distinguish between the areas that have blockers and those that -- and the area that does not?
A I was able to see the blockers, but I -- from the distance over the vacant part of the spent fuel pool, I did not look over the edge when I was closer, for fear of losing my hat.
Q So you noticed an area where there were blockers?
A Yes.
Q Was there an area that did not have blockers?

A Yes.
Q And did you notice how close they were to each other?
A Yes.
Q Could you tell us how close they were?
A Well, they were adjacent to each other.
Q How close?
A Probably a little larger than the width of a PWR fuel assembly.
Q Which would be what, approximately in inches?
~~A A little over eight inches maybe.~~
Q Could you describe for us what the blockers look like.
A I couldn't see -- I don't know whether they were just caps on top of a vacant storage container that prevented an assembly from being locked in there, or whether they were -- extended the whole length of the can. I just saw something blocking every other storage container.
Q And do you know what blockers are used for?
A Prevent fuel from inadvertently being loaded into the wrong location.
Q And did anything about the proximity of those two different areas that you have just observed give you cause for concern?

A No, not really. The blockers were pretty well visible from way on the other side of the pool.

Q Did you understand that one area was suitable for fresh fuel, as opposed to the other area, which was not suitable for fresh fuel?

A Yes.

Q And that these were -- these areas were separated by eight inches?

A Approximately, yes.

Q Let me ask you this: In terms of the ~~probability of the misplacement of fuel into the wrong~~ region, would that probability be lessened if they were a greater separation distance between those two regions, in your opinion?

A I'm not sure. I -- offhand, I don't see why. As I said, the region that had the blocking devices was fairly visible from clear on the other side of the spent fuel pool.

Q Well, would your answer be different if you knew that there has been, at Millstone, a series of fuel misplacements?

A I'd have to know what -- what type of misplacements and what the effect was.

Q But you're not willing to concede that having these two regions so close together has no potential effect on the probability of a misplacement that could not be diminished by greater separation?

A I suppose one could say that, but, you know, I -- I'm just not sure what -- you know, how much of a diminishment there would be or how much of an increase there would be in the probability having the regions next to each other.

Q You're not willing to concede any probability if the separation -- if one region were moved to the far end -- let's say the far end of the spent fuel pool, you're not willing to concede that that might make a difference in lowering the probability?

A I'm sure that it would. You're talking about a completely different area of the pool. If you're talking about just a few inches separation, I wouldn't see much of a difference.

Q Well, I didn't mention any number of inches in my question.

A That's why I'm confused as to what you're talking about.

Q Well, would you be willing then to concede that there would potentially be a reduction in the probability of a misplacement of fuel if those two regions were separated by a greater distance?

A Well, the thing that governs the placement of fuel is still the administrative procedures, and they are still -- I guess it would seem that if, you know, an area were removed quite a distance from the spent fuel -- the new fuel region, there could be somewhat less of a chance of a misloading, but still the loading is governed by administrative procedures and administrative controls, and those same procedures would have to be followed.

MS. BURTON: I have no further questions for this witness. Do you have anything?

MR. REPKA: I have just a couple of questions.

EXAMINATION BY MR. REPKA

Q Dr. Kopp, your position at the NRC, do you have responsibility for establishing the requirements for License Event Reports?

A No, I don't.

Q Do you have any particular knowledge of the standards for what types of events require an LER?

A No, I don't.

Q Do you ordinarily review and trend License Event Reports as part of your --

A Not as a matter of routine. If a particular event is given to me for review, then I would see it. Otherwise, I would not necessarily see them.

Q Is there an organization at the NRC that does that?

A There used to be. I'm not sure if it still exists or not. It was AEOD.

Q Does that function still exist somewhere, as far as you know?

A I don't know.

MR. REPKA: No further questions.

MS. BURTON: Attorney Hodgdon.

MS. HODGDON: Just -- I will have to look at my notes.

EXAMINATION BY MS. HODGDON

Q Could you explain more fully what your responsibilities are with relation to review of a particular licensee application for an amendment.

What do you do as a technical reviewer and what determination are you asked to make regarding that application; and if you could, the extent to which the use of administrative controls might come into that review?

A Boy.

Q That's a long question. I'm sorry. I'll break it up. You have done all of those. I'm just trying to get it together.

A Well, the first thing --

Q What are your responsibilities in your technical review?

A Reviewing the methods, the computer codes and the benchmarking to see whether the methods that are used are adequate to -- to predict the -- or to use for the analysis that's being performed; to look at the analysis itself to see if the results meet the staff's and the NRC's regulations as far as degree of subcriticality required, the events that should be looked at to ensure that that degree of subcriticality is met; and what was the rest of it?

Q The rest of the question was how do you focus on the use of administrative controls? I mean, is there any focus on administrative controls? Do you separate that out as being some box you check, or just how do you go about that? Is that something that leaps out at you, the words "administrative controls" as being something that you pay particular attention to, or --

A I guess the answer is no, not really.

Q What --

A I'm certainly aware of where administrative controls are used, but any time you move fuel or place fuel in a spent fuel pool, it's based on administrative controls.

MS. HODGDON: Thank you. I think that answers my question. I have no other questions.

MS. BURTON: I'm just going to go back to follow that for a moment.

FURTHER EXAMINATION BY MS. BURTON

Q Dr. Kopp, are you familiar with the circumstances at Millstone that led to the two-year shutdown of the entire station in 1996?

A No, I'm not.

Q Not at all familiar with that?

A No.

Q Are you familiar with the history of penalties and enforcement actions against that particular station?

A No, I'm not.

Q Are you at all familiar with a federal criminal investigation that led to $10 million in fines being imposed last September?

A No.

Q Northeast Utilities, you're not aware of that?

A No.

Q Are you aware of recent sanctions that were upheld against certain people at the station for retaliatory conduct against employees?

A No.

Q Are you aware of charges of felonies under the Atomic Energy Act that the company pleaded guilty to last fall?

A No, I'm not.

Q Falsification of training documents?

A No.

Q You never heard of any of this --

A This is not my -- I'm a nuclear engineer that looks at, primarily, criticality concerns in spent fuel pools. All those other issues are not in my -- I would not be involved in them.

Q Well, you have mentioned administrative controls. That really has to do with -- well, could you define "administrative control" for us.

A I would view it as a written procedure that is required to be followed performing a certain task.

Q And are you familiar with the performance at the Millstone station in terms of complying with administrative controls?

A No, I'm not.

Q You have no familiarity whatsoever?

A No, not anything that I could talk about or that I would be aware of enough to talk about.

Q Well, if you were aware that there was a history of failure over years to adhere to standards of administrative controls at that station, would that be an issue that would be of concern that you would need to analyze or should have analyzed in the course of preparing the work that you did in participating with this discovery?

A That would have been something that would have been handled by the NRC itself rather than my little area of the NRC.

Q So it would not inform your analysis at all?

A No.

MS. BURTON: I have nothing further.

MR. REPKA: Let me follow up with that, Dr. Kopp.

FURTHER EXAMINATION BY MR. REPKA

Q By saying that that wouldn't inform your analysis, do you mean to imply that the NRC would not look into those considerations?

A No, I would not.

Q You personally would not?

A As a personal reviewer. There are other parts of the NRC, I'm sure, that would.

Q Might that include the regional office?

A The regional office, enforcement, whatever.

Q The Office of Enforcement, the Commission itself?

A Yes.

Q And as far as you know, those organizations permitted Millstone to restart Unit 3 in 1998? Are you aware of that?

A Well, if they are operating, they must -- I guess they have.

MR. REPKA: Okay. Thank you.

MS. BURTON: Just to follow up.

FURTHER EXAMINATION BY MS. BURTON

Q But do you have any personal knowledge of the degree to which the regional office or other agencies within the NRC have evaluated issues of failure to comply with administrative controls?

A Millstone, no, I do not.

Q And are you aware or do you have familiarity with violations of administrative controls which may have occurred at the plant since the NRC approved restart of Units 2 and 3?

A No, I'm not.

Q Are you familiar with License Event Reports that may have been generated in this intervening time?

A No, I'm not, not unless I was specifically asked to review them.

MS. BURTON: I have nothing further for

this witness.

Anything further?

MR. REPKA: Nothing from me.

MS. BURTON: Attorney Hodgdon?

MS. HODGDON: No.

MS. BURTON: Thank you.

(Time noted 12:15 p.m.)

* * * * *

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of: : Docket No. 50-423-LA-3

:

Northeast Nuclear Energy Company :

:

Millstone Nuclear Power Station, Unit No. 3 : MAY 11, 2000

DEPOSITION OF LAURENCE T. KOPP, Ph.D.

____

LAURENCE T. KOPP, Ph.D.

Subscribed and sworn to before me this _____ day of ___________, 2000.

____

Notary Public

My Commission Expires:

STATE OF CONNECTICUT)
)
COUNTY OF NEW LONDON)

I, Kathryn Orofino, a Notary Public within and for the State of Connecticut, do hereby certify that I took the deposition of LAURENCE T. KOPP, Ph.D., a witness above-entitled action pursuant to 10 CFR Section 2.740a on the 11th day of May, 2000, at the Mystic-Noank Library, 40 Library Street, Mystic, Connecticut, at 10:15 a.m.

I further certify that said witness was by me duly sworn to testify to the truth, the whole truth and nothing but the truth, and that the testimony was taken by me stenographically and thereafter reduced to writing under my supervision; and that I am not an attorney, relative or employee of any party hereto nor otherwise interested in the event of this cause.

In witness whereof, I have hereunto set my hand and affixed my seal this 26th day of May 2000.

Kathryn Orofino
Shorthand Reporter #342
Notary Public

My Notary Public Commission Expires March 31st, 2001

EXHIBIT 28

NNECO's Supplementary Response to CCAM/CAM's Third Set of Interrogatories (June 21, 2000)

June 21, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	
	)	
Northeast Nuclear Energy Company	)	Docket No. 50–423–LA–3
	)	
(Millstone Nuclear Power Station,	)	
Unit No. 3)	)	ASLBP No. 00–771–01–LA

NORTHEAST NUCLEAR ENERGY COMPANY'S SUPPLEMENTARY RESPONSE TO CONNECTICUT COALITION AGAINST MILLSTONE AND LONG ISLAND COALITION AGAINST MILLSTONE'S THIRD SET OF INTERROGATORIES

Northeast Nuclear Energy Company ("NNECO") hereby files its supplementary response to the Connecticut Coalition Against Millstone ("CCAM") and the Long Island Coalition Against Millstone's ("CAM") (collectively, "Intervenors") "Third Set of Interrogatories and Requests for Production" ("Intervenors' Third Discovery Requests"),[1] which was served on NNECO on May 19, 2000.

I. Discovery Requests

A(4) Calculations of k_{eff}

(1) Given the implementation of the proposed re-racking of the Millstone 3 pool, and assuming an absence of soluble boron, what would be the calculated k-effective in each of the regions of the pool if various combinations of fresh fuel assemblies were placed in the racks? For this purpose, various combinations of fresh fuel assemblies would include one assembly, two adjacent assemblies, four adjacent assemblies, and a full rack, where in each case the surrounding cells would be occupied by assemblies of the highest reactivity allowed by the Technical Specifications.

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1 Although Intervenors refer to the subject request as their third, in reality it is their second.

NNECO's Response: In accordance with Atomic Safety and Licensing Board's Memorandum and Order (Discovery Rulings, 5/26/00 Telephone Conference), dated June 8, 2000, and NNECO's June 2, 2000, responses to the Intervenors' Third Discovery Requests, attached to this response are the assumptions and results for beyond-design-basis criticality calculations performed by Dr. Turner of Holtec International that NNECO will rely on in its written filing for the Subpart K proceeding.

Respectfully submitted,

Donald P. Ferraro

David A. Repka
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Lillian M. Cuoco
NORTHEAST UTILITIES SERVICE COMPANY
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ATTORNEYS FOR NORTHEAST NUCLEAR ENERGY COMPANY

Dated in Washington, D.C.
this 21st day of June 2000

TABLE 1

Criticality Calculations for Region 1

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.7611	k_{eff} well below critical
800 Technical Specification limit	Completely filled with fresh fuel of 5% enrichment	0.8916	Remains subcritical at Technical Specification limit of 800 ppm
0 Highly unlikely Loss of all soluble boron	Completely filled with fresh fuel of 5% enrichment	0.9728	Remains subcritical with system filled with fuel of maximum reactivity and concurrent loss of all soluble boron

* The k-effective values do not include bias and manufacturing tolerances, which are usually about 0.015Δk in Region 1.

TABLE 2

Criticality Calculations for Region 2

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.9384	Multiple accident condition remains sub-critical
2,000 Boron dilution	Completely filled with fresh fuel of 5% enrichment	0.9842	Minimum Boron concentration of 2000 ppm Boron to assure sub-criticality for multiple accident scenario
800 Technical Specification limit	8 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 2 rack	0.9794	Multiple accident with 8 fresh fuel assemblies remains sub-critical at Technical Specification limit of 800 ppm Boron
800 Technical Specification limit	5 assemblies fresh fuel of 5% enrichment mis-loaded into Region 2 otherwise filled with spent fuel	0.9663	Multiple accident with 5 fresh fuel assemblies remains sub-critical at Technical Specification limit of 800 ppm Boron
0 Loss of all soluble Boron	3 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 2 rack	0.9241	Maximum number of concurrent accidents in otherwise empty Region 2 with loss of all soluble Boron
0 Loss of all soluble Boron	1 assembly fresh fuel of 5% enrichment accidentally mis-loaded into Region 2 otherwise filled with spent fuel	0.9450	Single misplaced assembly accident with concurrent loss of all soluble boron

* k-effective values do not include bias and manufacturing tolerances which are usually about 0.01Δk for fresh fuel (Cases 1, 2, 3, and 5 above). For Cases 4 and 6 above, with spent fuel assemblies present in the Region 2 racks, the bias and uncertainties could be as large as 0.019Δk.

TABLE 3

Criticality Calculations for Region 3

ppm Boron	Fuel Array	k-effective*	Comment
2,600 Normal concentration	Completely filled with fresh fuel of 5% enrichment	0.8503	Multiple accident condition – remains sub-critical
1,320 Boron dilution	Completely filled with fresh fuel of 5% enrichment	0.9811	Minimum soluble Boron concentration of 1,320 ppm to assure sub-criticality with multiple accident scenario
800 Technical Specification limit	8 assemblies fresh fuel of 5% enrichment mis-loaded into otherwise empty Region 3 rack	0.9752	Maximum number of concurrent accidents in Region 3 at the Technical Specification limit of 800 ppm Boron
800 Technical Specification limit	5 assemblies fresh fuel of 5% enrichment mis-loaded into Region 3 otherwise filled with spent fuel	0.9528	Maximum number of concurrent accidents in Region 3 at the Technical Specification limit of 800 ppm Boron
0 Loss of all soluble Boron	1 assembly of fresh fuel 5% enrichment mis-loaded into Region 3 otherwise filled with spent fuel	0.9707**	Single misplaced assembly of the maximum reactivity with concurrent loss of all soluble Boron

* k-effective values listed do not include bias and uncertainties which are about 0.018Δk for fresh fuel (Cases 1, 2, and 3 above) and 0.029% Δk when the racks are otherwise filled with spent fuel (Cases 4 and 5 above).

** A single misloaded assembly accident remains sub-critical at nominal spent fuel pool water temperatures, including bias and maximum uncertainties. However, because the temperature coefficient of reactivity is positive for Region 3, should a concurrent abnormal increase in pool temperatures occur, Region 3 could potentially reach a critical condition in the absence of all soluble boron. At 150°F, as little as 30 ppm of soluble boron would ensure sub-criticality, including bias and uncertainties.

EXHIBIT 29

NRC Information Notice 94-13 (February 22, 1994)

EXHIBIT 30

AEC Press Release Entitled "AEC seeking public comment on proposed design criteria for nuclear power plant construction permits.") (November 22, 1965)

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)

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.

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 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

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

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.

EXHIBIT 31

Internal AEC Memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966) and attached Revised Draft of Generic Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966)(relevant excerpts)

CONTENTION TC-2: EXHIBIT 8

Internal AEC memorandum from G.A. Arlotto to J.J. DiNunno and Robert H. Bryan (October 7, 1966), and attached Revised Draft of General Design Criteria for Nuclear Power Plant Construction Permits (October 6, 1966)
(relevant excerpt)

Those Listed Below

October 7, 1966

G. A. Arlotto
Facilities Standards Branch, SS

REVISED DRAFT - GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

Attached is a revised draft of the General Design Criteria for Nuclear Power Plant Construction Permits dated October 6, 1966, which I developed for your consideration. In comparison with the previous draft, which was dated July 25, 1966, the attached version reflects the following:

1. Changes suggested by ACRS Subcommittee members at meetings of August 10 and September 21, 1966.

2. Changes suggested in the Backup Document dated August 9, 1966.

3. Changes suggested in memorandum from Robert H. Bryan to J. J. DiNunno dated October 3, 1966.

4. Changes resulting from discussions among the addressees and myself.

5. My suggestions which time did not permit resolution of with the addressees.

Attachment:
As Stated Above

Addressees:
J. J. DiNunno, Assistant Director for Reactor Standards, SS
Robert H. Bryan, Chief, Facilities Standards Branch, SS

OFFICE ▶	SS:FSB					
SURNAME ▶	Arlotto:jjb					
DATE ▶	10-7-66					

Revised Draft
10/6/66

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration. Category A criteria encompass critical safety areas so fundamental in the design, procurement, fabrication, and construction of the plant that modification for reasons of safety at the operating license review stage would be exceedingly difficult and costly; in essence, for practical purposes, decisions made at the construction permit stage in these areas are irrevocable. Where novel features

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are associated with criteria which are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain, they must be dealt with in a relatively complete way at the construction permit stage even if the "irrevocable" condition is not met. Category B criteria encompass safety areas where the modifications can be made for reasons of safety at the operating license review stage without placing an undue burden on the parties concerned. These criteria principally concerned with protecting the operational capability of the reactor may be dealt with in relatively less detail at the construction permit stage if more detailed information and analysis are not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

CRITERION 1 (Category A) QUALITY AND PERFORMANCE STANDARDS

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:

(a) 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 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 as necessary.

* A showing of sufficiency and applicability of standards used shall be required.

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(2) Active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

(3) A capability is provided to test periodically the delivery capability at a position as close to the spray nozzles as is practical.

(4) A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

CRITERION 10 (Category B) FUEL AND WASTE STORAGE SYSTEMS

Storage and handling systems for fuel and waste shall be designed on the basis that:

1. Possibilities for inadvertent criticality must be prevented by engineered systems or processes toevery extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

2. Reliable decay heat removal 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. 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.

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DATE ▶						

EXHIBIT 32

Letter from J.J. DiNunno, AEC, to David Okrent, ACRS (October 25, 1966) and attached October 20, 1966 Draft of General Design Criteria (relevant excerpts)

EXHIBIT 32

Letter from J.J. DiNunno, AEC, to David Okrent, ACRS (October 25, 1966) and attached October 20, 1966 Draft of General Design Criteria (relevant excerpts)

October 25, 1966

Dr. David Okrent, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D.C. 20545

Dear Dr. Okrent:

Enclosed for consideration of the ACRS are draft copies of the General Design Criteria for Nuclear Power Plant Construction Permits. This redrafted material includes a comparison of criteria contained in the Press Release dated November 22, 1965, and those contained in our latest draft dated October 20, 1966. In addition, we have included along with a revised draft of the criteria dated October 20, 1966, a comparison of the October 20 draft with the July 25 draft previously submitted and discussed with the ACRS Criteria Subcommittee.

Our October 20, 1966, draft attempts to reflect results of our last discussion with the ACRS Subcommittee, and we would like to have the scheduled November 9th meeting on criteria be based on the October 20th draft.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosures:
1. Rev. Draft dated 10/20/66 of General Design Criteria (18)
2. Comparison of Drafts dated 7/25/66 and 10/20/66 for General Design Criteria (18)
3. Comparison of Criteria in Press Release dated 11/22/65 and Those in Rev. Draft dated 10/20/66 (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
Clifford K. Beck, Deputy Dir. of Reg., w/encl.
Peter A. Morris, Director, DRL, w/encl. (6)
M. M. Mann, Asst. Dir. for Nuclear Safety, REG, w/encl.

OFFICE ▶	SS/ADIR					
SURNAME ▶	DiNunno:jjb					
DATE ▶	10-25-66					

REVISED DRAFT OF

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

October 20, 1966

The purpose of these criteria is to define or describe the basic safety objectives to be met in the design of a nuclear power plant. They are intended: (1) to serve as guidance to the applicant in preparing an application for an AEC construction permit and (2) to aid the AEC staff in reviewing that application.

The application of these criteria to a specific design involves a considerable amount of engineering judgment. There may be instances in which one or more of these criteria are unnecessary or are insufficient. It is not intended that the criteria be used as a check list of design objectives for all proposed plants, and the applicant is free to establish the safety of his design by alternative criteria. The criteria will be modified if, or as, future technological developments and experience warrant.

An applicant for a construction permit is expected to present a design approach together with data and analyses sufficient to give assurance that the design can reasonably be expected to fulfill all applicable criteria. It is recognized that the nature and detail of technical information and analysis required at the construction permit stage to provide such assurance may vary, depending on the particular criterion under consideration.

To provide guidance as to the relative emphasis expected at the construction permit stage, the criteria have been divided into two broad categories. Category A criteria involve aspects of facility design that are site-sensitive or are directly related to limiting the accidental release of radioactivity into the public domain. These aspects of facility design are also categorized by their marked influence on plans for construction and operation. From a practical viewpoint, aspects of facility design satisfying Category A criteria are relatively fixed at the construction permit stage and not amenable to change without serious disruptions of construction plans and incurrence of considerable costs. For these reasons, those aspects of facility design provided in fulfillment of Category A criteria must be dealt with in a relatively complete way at the construction permit stage.

Category B criteria are intended to reflect primarily those aspects of design that provide for safe operational control of the facility. Such features are generally less unique to a facility than those required for satisfying Category A criteria and are much less determinate of facility construction schedules. Modifications to such features that might prove necessary, for safety reasons, following issuance of a construction permit are much more likely to be accommodated without the pressures for compromise that might well accompany the more time-consuming and costly type changes. Under these circumstances, criteria principally concerned with the safe operational control of the reactor and designated as Category B may be dealt with in relatively less detail at the construction permit stage, if more detailed information is not available at that time.

All applicable safety criteria must, of course, be fulfilled as a condition for issuance of a license to operate the plant.

9.2.4.4 A capability is provided to test under conditions as close to the design as practical the full operational sequence that would bring the systems into action, including the transfer to alternate power sources.

FUEL AND WASTE STORAGE SYSTEMS

CRITERION 10 (Category B) FUEL AND WASTE STORAGE

10.0 Storage and handling systems for fuel and waste shall be designed on the basis that:

10.1 Possibilities for inadvertent criticality must be prevented by engineered systems or processes to every extent practicable. Such means as geometric safe spacing limits shall be emphasized over procedural controls.

10.2 Reliable decay heat removal 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. 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.

10.3 Shielding for radiation protection shall be provided as required from considerations of 10 CFR 20.

10.4 Containment of the systems shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

EXHIBIT 33

Letter from J.J. DiNunno, AEC, to Nunzio J. Palladino, ACRS (February 8, 1967), and attached Draft of General Design Criteria (relevant excerpts)

February 8, 1967

Mr. Nunzio J. Palladino, Chairman
Advisory Committee on Reactor Safeguards
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Palladino:

Enclosed for consideration by the Committee is a redraft of General Design Criteria. The format of the criteria has been changed. The subparts previously listed in earlier drafts have been made into separate criteria. The wording of these criteria is essentially the same as those in the October 20, 1966, draft, modified to reflect subsequent discussions held with the ACRS Subcommittee in November and recent developments of criteria for emergency core cooling systems.

An additional document showing the changes made from the last draft discussed with the ACRS is under preparation and will be forwarded by separate correspondence.

Sincerely yours,

J. J. DiNunno
Assistant Director for
Reactor Standards
Division of Safety Standards

Enclosure:
General Design Criteria for Nuclear
Power Plant Construction Permits (18)

bcc: Harold L. Price, Director of Regulation, w/encl.
Clifford K. Beck, Deputy Director of Regulation, w/encl.
M. M. Mann, Asst. Dir. for Nuclear Safety, w/encl.
C. L. Henderson, Asst. Dir. for Administration, w/encl.
Peter A. Morris, Director, DRL, w/encl. (6)
Edson G. Case, Deputy Director, DRL, w/encl.
Forrest Western, Director, DRL, w/encl.

OFFICE ▶	SS:ADIR	RL				
SURNAME ▶	DiNunno:jjb					
DATE ▶	2/8/67	2/8/67				

GENERAL DESIGN CRITERIA

FOR

NUCLEAR POWER PLANT CONSTRUCTION PERMITS

February 6, 1967

Table of Contents

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 61 - PREVENTION OF FUEL STORAGE CRITICALITY (Category B)

Possibilities for criticality in new and spent fuel storage shall be prevented by physical systems or processes to every extent practicable. Such means as favorable geometries 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 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 of 10 CFR 20.

CRITERION 64 - 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 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

EXHIBIT 34

Note by the Secretary, W.B.McCool, to AEC Commissioners re: Proposed Amendment to 10 CFR 50: General Design Criteria for Nuclear Power Plant Construction Permits (June 16, 1967)(relevant excerpts)

Rastro

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AEC-R 2/57

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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Chairman Seaborg	4	Asst. GM for Reactors	1
Commissioner Ramey	1	General Counsel	6
Commissioner Tape	2	Compliance	6
Commissioner Nabrit	2	Congr. Relations	2
Commissioner Johnson	2	Inspection	1
General Manager	2	Materials Licensing	2
Deputy Gen. Mgr.	1	Operational Safety	2
Dir. of Regulation	3	Plans & Reports	2
Deputy Dir. of Regulation	1	Public Information	2
Asst.Dir.of Reg.for Admin.	2	Reactor Dev. & Tech.	10
Asst.Dir.of Reg.for Reactors	1	Reactor Licensing	2
Asst. Gen. Mgr.	1	Reactor Standards	2
Exec. Asst. to GM	1	State & Lic. Relations	2
Asst. GM for Admin.	1	Chairman, AS&LBP	1

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[illegible] staff has considered all [illegible] in [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. <u>Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits," provides guidance for establishing the principal design criteria for nuclear power plants.</u>

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

APPENDIX A

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.
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

[illegible]

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 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.

EXHIBIT 35

Note by the Secretary, W.B.McCool, to AEC Commissioners re: Proposed Notice of Proposed Rulemaking for General Design Criteria, 32 Fed. Reg. 10,213 (July 11, 1967)

52.

Published 32 Federal Register 10213, July 11, 1967

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°32'42.5" 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 018° 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 350° 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

All

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[3]

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[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.

*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.]

EXHIBIT 36

Letter from William B. Cottrell, ORNL, to H.L.Price, AEC (September 6, 1967) and Enclosed ORNL Comments on Proposed GDC

DOCKET NUMBER PROPOSED RULE 50

OAK RIDGE NATIONAL LABORATORY

OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION

POST OFFICE BOX Y
OAK RIDGE, TENNESSEE 37830

DOCKETED USAEC SEP 15 1967 Office of the Secretary Public Proceedings Branch

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.

RECEIVED SEP 11 1967 U.S. AEC

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

<u>Title</u> - 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".

<u>Criterion 2</u> - Performance Standards

1. Line 7: Delete "performance" since this could be construed as applying to operating performance only.

2. In regard to earthquakes the "<u>appropriate margin</u> 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.

<u>Criterion 4</u> - 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.

<u>Criterion 5</u> - 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.

<u>Criterion 8</u> - 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 <u>two</u> 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 <u>much</u> greater than the worth of the most effective control rod is needed for reactors having many rods.

<u>Criterion 30</u> - 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.

<u>Criterion 31</u> - 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 <u>one</u> 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.

<u>Criterion 33</u> - 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.

<u>Section VII</u> - 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.

EXHIBIT 37

Letter from Edson G. Case, AEC, to Dr. Stephen H. Hanauer, ACRS (July 23, 1969), Enclosing General Design Criteria for Nuclear Power Plants (July 15, 1969) (relevant excerpts)

2 11/1

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

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 tha t 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.

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.

EXHIBIT 38

Memorandum from Edson G. Case, NRC, to Harold L. Price, et al., AEC, re: Revised General Design Criteria (October 12, 1970) and Enclosed Letter from Edward A. Wiggin, AIF, to Edson G. Case, NRC (October 6, 1970)

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

OCT 12 1970

Harold L. Price, Director of Regulation
Clifford K. Beck, Deputy Director of Regulation
Marvin M. Mann, Assistant Director of Regulation for Reactors
C. L. Henderson, Assistant Director of Regulation for Administration
S. H. Hanauer, Technical Advisor to the Director of Regulation
L. D. Low, Director, Division of Compliance
P. A. Morris, Director, Division of Reactor Licensing

REVISED GENERAL DESIGN CRITERIA

My memorandum of September 24, 1970, to Harold L. Price forwarded the latest revision of the General Design Criteria for your comments. Additions and changes to the June 4 version of the criteria were annotated.

Enclosed is a letter and enclosures which provide the AIF comments of the June 4 version of the criteria. Please note that the major Forum comments are discussed in the third enclosure to its October 6 letter. The revised criteria forwarded by my memorandum of September 24 appear to satisfy all of these major comments.

Please provide your comments on the revised criteria by Monday, October 19, so that review by the ACRS and final issuance of the criteria can be expedited.

Edson G. Case, Director
Division of Reactor Standards

Enclosure:
AIF Letter dated October 6, 1970,
to Edson G. Case w/encls
(except second enclosure)

cc: G. A. Arlotto, DRS

ATOMIC INDUSTRIAL FORUM INC.
475 PARK AVENUE SOUTH · NEW YORK, N.Y. 10016 · (212) 725-8300

October 6, 1970

Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D. C. 20545

Dear Ed:

The purpose of this letter and the enclosed material is to provide you with a commentary, developed by an ad hoc group convened under the aegis of the Forum's Committee on Reactor Safety, on the AEC-proposed "General Design Criteria for Nuclear Power Plants," as set forth in the AEC draft of June 4, 1970.

This commentary has been developed by, and represents the consensus view of, the following industry representatives, who have had an opportunity to participate either in redrafting and modifying the criteria or reviewing the same:

Robert D. Allen (Chairman) - Bechtel Corp.
Edwin A. Wiggin (Secretary) - Atomic Industrial Forum

Rennie Anderson - Combustion Engineering, Inc.
William Bley - Stone & Webster Engineering Corp.
Henry E. Bliss - Commonwealth Edison Co.
A. Philip Bray - General Electric Co.
Allan R. Collier - Westinghouse Electric Corp.
Walter D. Gilbert - General Electric Co.
Gilbert S. Keeley - Consumers Power Co.
Douglas V. Kelly - Pacific Gas & Electric Co.
William J. L. Kennedy - Stone & Webster Engineering Corp.
William Little - Babcock & Wilcox Co.
Lawrence E. Minnick - Yankee Atomic Electric Co.
James S. Moore - Westinghouse Electric Corp.
John N. Noble - Stone & Webster Engineering Corp.
Harold Oslick - Ebasco Services, Inc.
Warren H. Owen - Duke Power Co.

Rec'd Off. Dir. of Reg.
Date 10/12/70
Ti[illegible]

Richard F. Ranellone - General Electric Co.
William Smith - Babcock & Wilcox Co.
James E. Tribble - Yankee Atomic Electric Co.
Michael F. Valerino - Combustion Engineering, Inc.
Robert E. Wascher - Babcock & Wilcox Co.
John M. West - Combustion Engineering, Inc.
Robert A. Wiesemann - Westinghouse Electric Corp.

The enclosed material, which in its entirety comprises our commentary, includes the following five items:

1. A marked up version of the AEC draft of June 4 indicating the changes we believe should be incorporated prior to publication of the criteria.

2. A retyped version of the AEC draft of June 4 incorporating the changes referred to above.

3. A discussion of the major changes recommended. Our consensus agreement with the criteria as modified is dependent upon their acceptance.

4. An explanation of certain detailed changes which we believe to be both necessary and desirable if the criteria are to prove of maximum usefulness to the AEC and the industry. Omitted from this listing are minor changes, for the most part self-explanatory, which have been suggested in the interest of enhancing the clarity of certain criteria but which do not alter either their scope or intent.

5. An excerpt which we believe should be incorporated in the Statement of Considerations at the time the criteria are published.

We wish to emphasize the importance attached to the concerns underlying the major changes recommended. We very much hope that these concerns can be accommodated by adoption of the recommended changes or in some other equally satisfactory manner.

Submission of this consensus commentary is not intended to preclude the subsequent submission of individual comments by those named above or by other industry representatives, once the criteria have been published. Conversely, it is not expected that the group named above or the Forum Committee on Reactor Safety would wish to offer further

ATOMIC INDUSTRIAL FORUM INC.

comments if the recommendations set forth in this commentary are adopted.

Please let us know if you desire further clarification of these comments. Also, should you wish further elaboration of the comments, we would be pleased to convene a representative group of those named above to meet with you and your associates.

We appreciate the opportunity to comment on this important document.

Sincerely,

Edwin A. Wiggin

EAW:erk
Enc.

DRAF

GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

June 4, 1970

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. 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 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 types of nuclear power units.

Accordingly, these General Design Criteria are intended to reflect current licensing review practice.

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 of accidents which could cause undue risk to the health and safety of the public. There will be some water-cooled nuclear power units for which these General Design Criteria are not sufficient for this purpose, and additional criteria must be identified and satisfied by the design in the interest of public safety. It is expected that additional or different criteria ~~will~~ may 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 units such as these, departures from the General Design Criteria must be identified and justified.

Insert (:)-see next page

The requirements of these General Design Criteria shall be supplemented or modified as necessary to cope with the existence or consequences of a previously unidentified physical condition important to safety. The effective date for the application of industry codes and standards shall be as specified in Title 10 of the Code of Federal Regulations.

Insert (1)

The development of these General Design Criteria is not yet complete. For example, some of the definitions need further amplification. Also, certain of the specific design requirements for structures, systems, and components important to safety have not as yet been suitably defined so that they can be generalized as criteria. For these reasons it is expected that the criteria will be augmented and revised from time to time as important new or changed requirements such as these are identified and developed.

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 prevent or mitigate the consequences of accidents which could cause 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 system used for normal reactor coolant makeup, ~~system~~ from ~~any size~~ breaks in the ~~piping, pressure vessels, pumps, and valves connected to the reactor pressure vessel and which are part~~ of 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.[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. Mechanical 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 ~~any~~ selected passive components (assuming active components function properly), results in a

[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 as a general design criterion.

loss of the capability of the system to perform its safety functions.[2] ~~The failure of a passive component need not be considered in the design of mechanical systems if it can be demonstrated that the design is acceptable on some other defined basis, such as an appropriate combination of unusually high quality, high strength or low stress, inspectability, repairability, or short-term use.~~

<u>ANTICIPATED OPERATIONAL OCCURRENCES</u>

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 the recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

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 mechanical system should be considered in designing the system against a single failure are under development as a general design criterion.

CRITERIA

I. OVERALL REQUIREMENTS

CRITERION [illegible] - 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, if required by unusual design or site characteristics, 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, as defined in 10 CFR Part 50, Appendix B, 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, (2) sufficient 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 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 and testing, and postulated accidents. These structures, systems, and components shall be

to the extent necessary
appropriately protected/~~against~~ dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from/ the effects of events and conditions ~~sources~~ outside the nuclear power unit.

~~CRITERION 5 - PROTECTION AGAINST INDUSTRIAL SABOTAGE~~

~~Structures, systems, and components important to safety shall be physically protected to minimize, consistent with other safety requirements, the probability and effects of industrial sabotage.~~

CRITERION 6 - 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 ~~design~~ limits are not exceeded during all conditions 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.

CONTROL
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 ~~of~~ specified acceptable fuel design limits are not possible or can be reliably and readily detected and ~~suppressed.~~ controlled.

CRITERION 13 - REACTOR INSTRUMENTATION AND CONTROL

Instrumentation and control shall be provided to monitor and to maintain variables within prescribed operating ranges, including those variables and systems which can affect the fission process and the integrity of the reactor core.

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 all conditions 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 onsite and offsite power systems shall each ~~The safety function for each system alone shall be~~ ~~to~~ provide sufficient capacity and capability to assure that (1) specified acceptable fuel damage ~~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.

Element of system important to safety
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 supplie
~~Two physically independent transmission lines, each with the capability~~
by two transmission lines designed and located so as to suitably minimize the likelihood of
~~of supplying electrical power from the transmission network to the switch-~~
their simultaneous failure under operating, accident and expected environmental conditions.
~~yard, and two physically independent circuits from the switchyard to the~~
Two physically independent circuits from the switchyard to the onsite electrical distributi
~~onsite electrical distribution system shall be provided.~~ Each of these
system shall be provided.
circuits shall be designed to be available in sufficient time following a
loss of ~~electrical power from all other alternating current sources, including~~
power , in the absence of a loss-of-coolant accident,
all /onsite electrical/sources/ to assure/that specified acceptable fuel design
limits and design conditions of the reactor coolant pressure boundary are
Assuming a loss-of-coolant accident, from the switchyard to the onsite electrical
power distribution system shall
not exceeded./ One of these circuits ~~shall be designed to~~ be available
in sufficient time
~~immediately following a loss of coolant accident~~ to assure that core cooling,
important to safety
containment integrity, and other ~~vital safety~~ functions/are maintained.

Provisions shall be included to minimize the probability of losing
from the offsite electrical power system sources
electrical power/ ~~via any of the remaining circuits~~ 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 ~~required for~~ 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 active components of the systems, such as onsite emergency 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, although not necessarily while the plant is at power, the full operational sequence that brings the systems into operation, including operation of the initiation logic ~~protection system~~, and the required transfer of power among the nuclear power unit, the offsite power system, and the onsite emergency 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 emergency 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 damage ~~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. 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.

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 functional performance 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. ~~in the event of systematic, nonrandom, 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 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 absure that safety is not significantly impaired, ~~considering the possibility of systematic, nonrandom, concurrent failures of control system components or channels, or of those common to the control and protection systems~~.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

specified

The protection system shall be designed to assure that/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 CAPABILIT

Two independent reactivity control systems, ~~preferably~~ of different

shall be provided.

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 failure of the highest worth rod to insert, ~~appropriate margin for malfunctions such as stuck rods~~, specified acceptable fuel damage ~~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 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, including consideration of any rods failing to insert as a consequence of the accident.

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. Their design shall reflect consideration of systematic, nonrandom, concurrent failures of redundant elements.~~

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 in accordance with applicable industry codes. ~~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 under stressed 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

Components which are part of the reactor coolant pressure boundary shall be designed to permit/(1) periodic inspection and testing of important areas in accordance with applicable industry codes 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 and for offsite electrical power system operation 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, suitable interconnections, and leak detection and isolation capabilities shall be provide to assure that for either onsite or ~~and for~~ offsite electrical power system operation 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 a ~~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. ~~The performance of the system shall be evaluated conservatively.~~

Suitable redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for either onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 36 - INSPECTION AND PRESSURE TESTING OF EMERGENCY CORE COOLING SYSTEM ~~COMPONENTS~~

~~Components of~~ The emergency core cooling system shall be designed to permit periodic inspection and appropriate pressure testing of important components ~~areas and features, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping,~~ to assure their structural and leaktight integrity as a measure of and the full design capability of the system.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2) to the extent practical, the operability of the system as a whole, ~~and, under conditions as close to design as practical~~. The full operational sequence that brings the system into operation, including operation of the initiation logic ~~protection system,~~ the transfer between normal and emergency power sources, and 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 a ~~any~~ loss-of-coolant accident and maintain them at acceptable ~~low~~ levels.

Suitable redundancy in components and features, suitable interconnections, and leak detection, isolation, and containment capabilities shall be provided to assure that for either onsite ~~and for~~ or offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 39 - INSPECTION AND PRESSURE TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM ~~COMPONENTS~~

~~Components of~~ The containment heat removal system shall be designed, insofar as practical, to permit periodic inspection and appropriate pressure testing of important components ~~areas and features, such as the torus, pumps, spray nozzles, and piping~~. to assure their structural and leaktight integrity as a measure of ~~and~~ the full design capability of the system.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves~~ and (2) to the extent practical the operability of the system as a whole, and, under conditions as close ~~to the design as practical,~~ the full operational sequence that brings the system into operation, including operation of the initiation logic ~~protection system~~, the transfer between normal and emergency power sources, and operation of the associated cooling water system.

CRITERION 41 - CONTROL OF 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 limit ~~reduce~~, consistent with the functioning of other associated systems, the release ~~concentration and quantity~~ of fission products ~~released~~ to the environment following postulated accidents, such that acceptable limits are not exceeded, 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, suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for either onsite ~~and for~~ or offsite electrical power system operation its safety function can be accomplished assuming a single failure.

CRITERION 42 - INSPECTION AND PRESSURE TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS ~~COMPONENTS~~

~~Components of~~ The containment atmosphere cleanup systems shall be designed, insofar as practical, to permit periodic inspection and appropriate pressure testing of important components ~~areas and features such as filter frames, ducts, and piping~~ to assure their structural and leaktight integrity as a measure of ~~and~~ the full design capability of the systems.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the systems ~~such as fans, filters, dampers, pumps and valves~~ and (2) to the extent practical, 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 the initiation logic, ~~protection system,~~ the transfer between normal and emergency power sources, and 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 or ~~and~~ accident conditions.

Suitable redundancy in components and features, suitable interconnections, and leak detection and isolation capabilities shall be provided to assure that for either onsite or ~~and for~~ offsite electrical power system operation the system safety function can be accomplished assuming a single failure.

CRITERION 45 - INSPECTION AND PRESSURE TESTING OF COOLING WATER SYSTEM ~~COMPONENTS~~

~~Components of~~ The cooling water system shall be designed insofar as practical to permit periodic inspection and appropriate pressure testing of important components ~~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 to permit periodic functional testing which will provide a measure of (1) the operability and performance of the active components of the system, ~~such as pumps and valves,~~ and (2) to the extent practical the operability of the system as a whole, ~~and, under conditions as close to design as practical~~ the full operational sequence that brings the system into operation for reactor shutdown and ~~for~~ loss-of-coolant accidents, including operation of the initiation logic ~~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 allowable ~~design~~ leakage rate, ~~and, with sufficient margin,~~ the calculated pressure and temperature conditions resulting from a ~~any~~ loss-of-coolant accident. The design ~~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 ~~necessarily~~ be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at pressures up to and, if necessary, including the ~~containment~~ design pressure.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit insofar as practical (1) visual inspection of all ~~important areas, such as~~ penetrations, (2) an appropriate materials surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows ~~at containment design pressure~~.

INSERT (2) - see next page

~~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.~~

INSERT (2)

CRITERION 54 - PROVISIONS FOR CONTAINMENT ISOLATION

Piping which penetrates the containment must be provided with two isolation barriers; one barrier must be located outside the containment and one must be inside the containment, unless it can be demonstrated that the design is acceptable on some other defined basis.

The definition of an isolation barrier is either a suitably designed closed system trip valve, check valve or a manually closed valve under administrative control.

Using this definition four general classifications are derived:

1. Two closed systems - one inside, one outside, no isolation valves required.
2. No closed systems - one valve inside and one valve outside required.
3. Closed system inside - no valve inside, valve required outside.
4. Closed system outside - no valve outside, valve required inside.

NOTE 1: The same criteria apply to lines which are used after an accident except that manual isolation is acceptable and in the case of instrument lines, a check valve or manual valve inside or outside containment is acceptable.

NOTE 2: An isolation valve outside containment shall be located as close to to the containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

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 automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability o 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 hi ner quality in design, fabrication, and testing, additional provision 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 PRESSURE BOUNDARY ISOLATION VALVES

Each line which connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with one automatic isolation valve inside and one automatic isolation valve, other than a simple check valve, outside of containment, unless it can be demonstrated that the design is acceptable on some other defined basis. The valve outside of containment shall be located as close to containment as practical and upon loss of actuating power the automatic isolation valves shall be designed to take the position that provides greater safety.

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, other than a simple check valve. This valve shall be outside of containment and shall be located as close to containment as practical.

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 ~~maintain suitable~~ control the handling and release of ~~over~~ radioactive materials in gaseous and liquid effluents and in solid wastes produced during normal reactor operation, including anticipated operational occurrences, within acceptable limits. 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 their release to the environment.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVE WASTE SYSTEMS ~~RADIOACTIVITY CONTROL~~

The fuel storage and handling and radioactive waste systems ~~and other systems which may contain radioactivity~~ shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be provided with ~~designed~~ (1) with a capability to permit inspection and testing of ~~important areas and features of the~~ components important to safety ~~of these systems~~, (2) with suitable shielding for radiation protection, (3) ~~with~~ appropriate ~~containment~~, confinement, and filtering systems, and (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 designed ~~(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

Instrumentation ~~Appropriate systems~~ shall be provided in fuel storage and radioactive waste systems and associated handling areas ~~(1)~~ to detect and alarm any 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.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at ______________________ this ____________

day of ______________ 1970.

For the Atomic Energy Commission

____

W. B. McCool
Secretary

A Discussion of Major Changes Recommended

There are a number of criteria which as drafted cannot be accepted by the industry for one or more of the following reasons: (1) it represents an unnecessary and unjustified escalation of licensing requirements, (2) there is no clear or common understanding on the part of the AEC and the licensee as to what it would take to meet the requirement, and (3) it is premature to attempt to incorporate the requirement into general design criteria inasmuch as the technical rationale for the requirement has not been fully developed.

Loss-of-Coolant Accident

The definition of the loss-of-coolant accident as set forth in the AEC draft of June 4 clearly represents an escalation of licensing requirements inasmuch as it refers to "any size break" in the "pressure vessels, pumps, and valves connected to the reactor pressure vessel" as well as to a break in the piping. These additional breaks should not be postulated by license reviewers and certainly should not be incorporated into general design criteria in the absence of a realistic technical rationale, the basis for which can be developed only through further study. That study is now being undertaken by an ACRS subcommittee and by an ad hoc Forum group.

Single Failure

As the definition of "single failure" appears in the AEC draft of June 4, it postulates the failure of passive components in both mechanical and electrical systems. Although current licensing review practice assumes the failure of passive components in electrical systems, the extension of the general concept to mechanical systems represents an escalation of licensing requirements for which no technical rationale has been developed. Further, the definition leaves open ended the number and type of mechanical systems to which it could be applied. Indeed, an undisciplined application of the definition would presumably lead to postulating such failures as to make it impossible to design operable systems. Clearly, a single failure concept which would permit the indiscriminate application of postulated failures of passive components in mechanical systems should not be incorporated into general design criteria.

Industrial Sabotage

The AEC draft of June 4 includes as Criterion 5 "Protection Against Industrial Sabotage" which reads" "Structures, systems, and components important to safety shall be physically protected to minimize, consistent with other safety requirements, the probability and effects of industrial sabotage."

Policy considerations involved in the proposed requirement are of such significance that a direct discussion of top utility management personnel with members of the Commission would appear to be prerequisite to resolution of the issues that would be raised in implementing the proposed criterion.

Transmission of Offsite Electrical Power

Criterion 17, "Electrical Power Systems," as it appears in the June 4 draft, includes the requirement: "Two physically independent transmission lines, each with the capability of supplying electrical power from the transmission network to the switchyard, and two physically independent circuits from the switchyard to the onsite electrical distribution system shall be provided."

A literal interpretation of this requirement would call for two transmission lines mounted on different sets of towers located on different rights-of-way. Not only is this an unwarranted escalation of licensing requirements, but for many sites the requirement would neither be desirable nor possible to meet. Further, such a requirement would be contradictory in many instances with requirements being imposed on licensees by environmental considerations.

License applicants should be permitted the option of satisfying the integrity of emergency offsite electrical power service by means other than would be permitted by the criterion as now drafted.

Systematic, Nonrandom, Concurrent Failures of Redundant Elements

Criteria 22, 24 and 29, as set forth in the AEC draft of June 4, all deal with protection and reactivity control systems and all postulate "systematic, nonrandom, concurrent failures of redundant elements." This postulated failure mode is not acceptable to the industry for the following reasons: (1) there is no indication of what requirements are involved, (2) it would provide a "hunting license" for an undisciplined imposition of requirements, (3) there is no logical basis for limiting the concept to protection and reactivity control systems, and (4) the reactor systems suppliers are only now in the early stages of studies which the AEC regulatory staff has asked them to undertake in this area.

Until such time as the requirements which would be imposed by this postulated failure mode can be clearly defined and supported by sound technical rationale, they should not be incorporated into general design criteria.

Containment Isolation

Criterion 54 through Criterion 57, as set forth in the AEC draft of June 4, provide a number of requirements dealing with containment isolation. As drafted, some of these requirements are difficult to interpret and appear to represent an escalation of current licensing practice. Informal discussions with the AEC regulatory staff have not proved successful in developing a mutually satisfactory format for these criteria.

EXHIBIT 39

Final Rule, General Design Criteria for Nuclear Power Plants, 36 Fed. Reg. 3,255 (February 20, 1971)

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. 71-520]

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.[2] 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] General design criteria for chemical processing facilities are being developed.

2. 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:

(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.

[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 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.

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.

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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