January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

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

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

VOLUME 7

EXHIBIT 12 - 20

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

DEPOSITION OF: LAURENCE I. KOPP, Ph.D.

DATE: November 4, 1999
Commencing at 2:15 p.m.

PLACE: Goya Conference Room
Four Points Sheraton Hotel
37611 U.S. Highway 19 North
Palm Harbor, Florida 34684

REPORTED BY: Dale E. DeFranco, RPR
Notary Public
State of Florida at large

COPY

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APPEARANCES:

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* E X H I B I T S *

Exhibit No.	Description	Page
7	Resume of Laurence I. Kopp, Ph.D.	5
8	Proposed Revision 2 to Regulatory Guide 1.13	18
9	August '98 memo from Laurence Kopp	21
10	October 25, '96 letter from Tim Collins	36

14

1 DR. LAURENCE I. KOPP,
2 the deponent herein, being first duly sworn, was examined
3 and testified as follows:
4 DIRECT EXAMINATION
5 BY MS. CURRAN:
6 Q. Good afternoon, Dr. Kopp.
7 A. Good afternoon.
8 Q. Have you been deposed before?
9 A. No, I haven't.
10 Q. I'm going to ask you questions regarding your
11 involvement in the review of the Harris Spent Fuel Pool
12 Expansion License Amendment Application.
13 A. Uh-huh.
14 Q. And your experiences regarding criticality
15 analysis in general. I don't mean for my questions to be
16 confusing and if there's any question that you don't
17 understand please ask me for clarification and I'll do my
18 best to clarify it.
19 If for any reason you need a break just ask for
20 a break and you're welcome to one. You're under --
21 you're aware you're under oath --
22 A. Yes.
23 Q. -- in this deposition. Okay.
24 MS. CURRAN: I'd like to ask the court reporter
25 to mark as Exhibit 7 the resume of Laurence,

14

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

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

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

A. Yes.

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

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

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

A. Oh, probably eight or ten years.

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

A. No.

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

A. In spent fuel pools?

Q. Yes.

A. Primarily there are some others and there are a

14

1 few that are being trained in case I ever decide to
2 retire.
3 Q. Would you say that you are familiar with all of
4 the criticality analyses for spent fuel pools that are
5 submitted to the NRC?
6 A. I would say yes, the ones that have been
7 submitted since the early 1980ies. I think 1983, when I
8 first started working on criticality analysis, spent fuel
9 pools.
10 Q. Before 1983 were you involved in criticality
11 analysis at all?
12 A. Not for spent fuel pools, no.
13 Q. Are you the person who is presently responsible
14 for review of the criticality issues raised by the Harris
15 License Amendment Application with respect to Pools C and
16 D?
17 A. Yes.
18 Q. Now, the NRC staff issued request for
19 additional information in April of 1999 which was
20 responded to by CP&L in June of 1999 regarding
21 criticality issues; is that correct?
22 A. I'm not sure of the dates, but yes, it requests
23 for additional information around that time frame.
24 Q. And it was answered, right?
25 A. Yes.

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

A. Essentially it is, yes.

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

A. Yes.

Q. And what is it?

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

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

A. Yes. Somewhere around there.

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

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

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

A. Not very much.

Q. But somewhat?

A. Somewhat.

Q. I'm going to ask you some questions about that and if you don't know just tell me. Is it correct to say that around the time of issuance of General Design Criterion 62 to 10 CFR Part 50, Appendix A, there were no

license tec specs requiring boron to be present in fuel pool cooling water?

A. Could you restate that.

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

A. Uh-huh.

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

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

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

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

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

15

1 A. That's right.
2 Q. Do you know when was the first time that such a
3 requirement was entered into the tec specs for any
4 nuclear plant?
5 MS. UTTAL: By such a requirement --
6 Q. For soluble boron and spent fuel pools.
7 A. I don't recall.
8 Q. You haven't hazarded a guess or a arange?
9 A. As far as, as long as I remember it it's been
10 in most of the tecnical specifications.
11 Q. But your memory goes back to, say, the early
12 '80ies.
13 A. For that specification, yes.
14 Q. So why is Harris different?
15 A. Well, it has been in some specs and it has not
16 been others. There was no requirement to have it in all
17 the technical specifications up until a few years ago
18 when we developed the improved technical specifications.
19 Q. What year was that?
20 A. I'd say about, between three to five years ago.
21 Q. Were the improved technical specifications --
22 A. Can I clarify?
23 Q. Sure.
24 A. I don't know if I said it was required in all
25 technical specifications in the past. It had been in

15

1 many technical specifications in the past but it was not
2 a requirement as far as I know.
3 Q. Okay. Would this be a requirement in, when you
4 talk about the new and improved technical specifications,
5 would you be referring to New Reg 1431, the Standard
6 Technical Specifications for Westingtonhouse Plants?
7 A. Yes.
8 Q. Now, my copy of that is dated September 1992.
9 That's for Volume, it just says Volume I. It doesn't
10 give a revision number. Revision Zero. Would that seem
11 to be --
12 MR. HOLLAWAY: Do you want to take a look at
13 that?
14 Q. I don't have the whole thing but you're welcome
15 to look at it, the first page.
16 A. Yes. This is it. This is what I'm referring
17 to.
18 Q. So that would have been around 1992.
19 A. Has it that long ago?
20 Q. Time flies, doesn't it? So just to make sure I
21 understand, it was with the promulgation of these
22 standard tec specs that it became an NRC requirement to
23 include a provision for soluble boron in spent fuel pools
24 in the tec specs.
25 A. For those plants that adopted the improved

15

1 standard tec specs and eliminated their older tec specs.
2 Q. And in what proportions of plants did that?
3 A. Well, it's a continuing basis. They're coming
4 in gradually. Some of them are adopting the improved tec
5 specs and others are not. I don't know what percentage
6 have adopted them and what percentage have not adopted
7 them.
8 Q. Is it the NRC's goal that most or all plants
9 will adopt these?
10 A. I don't know if it's our goal, but it's a joint
11 effort by NRC and industry to simply tec specs and make
12 them more consistent among different venders.
13 Q. Would it be correct to say that at the time of
14 the promulgation of GDC-62 or thereabouts, which was in
15 1971, there were no technical specifications in nuclear
16 power plant licenses that imposed burnup limits on fuel
17 to be stored in fuel pools?
18 A. As far as I recall that's right, there were
19 none.
20 Q. When was a burnup limit on fuel stored in pools
21 first inserted in a set of nuclear power plant tec specs?
22 A. I'd have to be guessing, but I will say the
23 early 1980ies, maybe 1981, '82.
24 Q. Do you remember what the plant was?
25 A. No, I don't.

Q. Do you remember what the circumstances were?

A. I'm not sure.

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

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

Q. And the NRC approved the concept?

A. Yes.

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

A. I'm not sure.

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

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

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

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

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

A. The NRC, no, no.

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

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

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

A. Not that I know of, no.

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

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

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

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

1 assemblies in Pools C or D?
2 MS. UTTAL: Objection. That's a compound
3 question. You're asking him whether the NRC
4 has done analysis to predict the probability or I
5 assume an analysis of the, how something like that
6 would happen? Are those the two things you're
7 asking?
8 Q. To identify possible scenarios and to predict
9 their probability. Or to predict their probability?
10 A. No. We haven't done either of those. If I may
11 say, we assumed the probability of misplacing the fuel
12 assemblies, one, that's why we required the analysis be
13 done, for 100 percent probability to misplace the fuel
14 assembly. That was the basis for our request for
15 additional information on that analysis.
16 Q. Has the NRC done any evaluation of the set of
17 scenarios by which through a single error CP&L might
18 place more than one out-of-compliance fuel assembly in
19 Pools C or D?
20 A. No.
21 Q. Has the NRC requested CP&L to perform any such
22 analysis?
23 A. No.
24 Q. I'm going to ask you a question that may sound
25 familiar. I asked Dr. Turner the same thing. I'd like

1 to ask some questions regarding how you, as a
2 professional criticality analyst, would evaluate the
3 envelope of criticality events -- I'm sorry, to identify
4 the envelope of criticality events that could occur at
5 Harris.
6 I'd like you to consider a hypothetical problem
7 in criticality analysis for fuel in pools. In this
8 problem the physical configuration of the racks is fixed.
9 The variables are, one, soluble boron and, two, the
10 combined burnup, slash, enrichment of the fuel. In this
11 problem some number of fuel assemblies may exceed
12 acceptable burnup, splash, enrichment levels.
13 In addition boron concentration may be anywhere
14 from zero to 2000 PPM. Your task in this problem is to
15 identify the set of scenarios involving combinations of
16 parameters, one, which is soluble boron and, two, which
17 is the combined burnup, slash, enrichment of the fuel.
18 Such that criticality just occurs. I.e., k effective
19 equals one.
20 For the purposes of this problem I'd like you
21 to define that set of scenarios as the envelope of
22 criticality events for this pool.
23 A. Okay. First of all may I say that if the fuel
24 assembly exceeds the burnup enrichment limits it's safer
25 that it was below the burnup enrichment limits. The

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1 curve is a curve that requires burnup to either meet that
2 limit or exceed it. Not be below it. So if your burnup
3 is higher than the tec spec limit on the burnup you're
4 safer, you're fuel assembly is less reactive.
5 Q. And the problem I'm posing to you within the
6 envelope the fuel will be more reactive; outside the
7 envelope the fuel will be less reactive. Do you
8 understand?
9 A. Right.
10 Q. Using this hypothetical how would one determine
11 the envelope of scenarios involved in criticality?
17
12 A. We base it on NRC requirements which require
13 criticality not to be reached but to maintain the
14 five-percent subcriticality margin at all times, even for
15 the worst conceivable accident which would be a loss of
16 all the boron in pool water, somehow diluted the pool
17 from 2000 PPM down to zero, the calculations showed
18 you're still at least five-percent subcritical. And as
19 far as an envelope of calculations that seems to me to be
20 the bounding point.
21 Q. But I've asked you to look at a combination of
22 events involving two factors.
23 A. Right.
24 Q. We have factor one, which is soluble boron and
25 factor two, the combined burnup enrichment of the fuel.

1 So I'm asking you to look at combinations of these
2 factors. And in looking at those combinations I would
3 assume there are various combinations you would look at,
4 how would you determine the envelope?
5 MS. UTTAL: Objection. He's already stated
6 that the envelope or the bounding, the boundary
7 is the loss of boron. I don't know what purpose is
8 served by talking about things that are inside the
9 boundaries, or he stated what the outside boundary
10 is.
11 A. The burnup curves are based on a five-percent
12 subcriticality margin assuming no credit for boron. One
13 never goes through some type of analysis where you would
14 be critical, so one never does calculate a k effective
15 1.0. We always have a five-percent safety margin.
16 Q. But what you just said to me assumes that
17 there's only one misplaced fuel assembly, right?
18 A. No. What I said assumes that we have the
19 maximum reactivity accident possible. Loss of all the
20 boron in the pool water.
21 Q. Is that your professional answer as a scientist
22 to the question?
23 A. Yes.
24 Q. One of the pieces of guidance that the NRC uses
25 to evaluate criticality analyses is proposed Revision 2

17

1 to Reg Guide 1.13 which is dated December 1981; is that
2 correct?
3 Would you like me to show you that document?
4 A. I know what document you're talking about.
5 That was never officially issued as a reg guide and
6 therefore it's hard to say that it's something that the
7 commission would rely on.
8 Q. Well, I'm asking what the staff relies on.
9 Does the staff rely on it?
10 A. I do not rely on it. I know what's in there,
11 but I know some of things in there are not, are obsolete
12 and I know some of the things in there have been updated
13 since then.
14 Q. In terms of criticality analysis. Well, why
15 don't I pass this out and ask the reporter to mark as
16 Exhibit 8, Proposed Revision 2 to Regulatory Guide 1.13,
17 entitled Spent Fuel Storage Facility Design Basis. It's
18 dated December 1981.
19 (Whereupon, Exhibit Number 8 was
20 marked for identification.)
21 Q. Dr. Kopp, this document that I'm showing you,
22 is this a copy of the guidance document we've been
23 talking about?
24 A. It's a copy of Proposed Reg Guide 1.13, yes.
25 Q. Okay. Now as you were saying a little earlier

1 that is still in proposed or draft form, right?
2 A. I don't know what's happened to it now. It's
3 almost 20 years and I have not heard anything about it.
4 Q. You haven't heard any rumors that it's about to
5 come out?
6 A. No.
7 Q. We're all waiting. Can you tell me what
8 measures or what aspects of the guidance that relate to
9 criticality control have been changed or updated?
10 A. As far as I see it's been updated to reflect
11 the recent position that we have granted in partial
12 credit for soluble boron for normal conditions.
13 Q. And in what case was that granted?
14 A. It was a generic topical report from
15 Westinghouse that we reviewed and approved three years
16 ago. Somewhere in that time frame.
17 Q. So that's the one thing that you can offer that
18 is changed since this --
19 A. Yes.
20 Q. -- draft or reg guide was published?
21 A. From quickly glancing at it, yes.
22 Q. Okay. Dr. Kopp, you were here for the
23 deposition of Dr. Turner; is that correct?
24 A. Yes.
25 Q. Do you recall a discussion about an

17

1 unsubstantiated rumor that the NRC is having second
2 thoughts and may rescind it's approval of partial boron
3 credit?
4 A. Yes.
5 Q. Is there any truth to that rumor?
6 A. Not that I know of.
7 Q. Do you know where that rumor may have come
8 from?
9 A. No, I don't.
10 Q. You have no idea what would have started it?
11 A. No. We have granted partial boron credit for
12 several plants already.
13 Q. Which plants are those?
14 A. Prairie Island was the first. I believe Vogtle
18
15 was another one. V-o-g-t-l-e.
16 Q. Was Comanche Peak another one?
17 A. I'm not sure if it was Comanche Peak or South
18 Texas. It might have been both of those. Some have not,
19 the amendment has not officially gone out yet.
20 St. Lucy II was also another. I'm not sure
21 which of these have already been officially approved and
22 which were just approved by our branch as far as
23 criticality goes.
24 MS. CURRAN: I'd like to as the court reporter
25 to mark as Exhibit 9 an August 1998 memorandum from

1 Laurence Kopp, Senior Reactor Engineer, Reactor
2 System Branch, Division of Systems Safety and
3 Analysis to Timothy Collins, Chief Reactor
4 Systems Branch, Division of Systems Safety and
5 Analysis.
6 Attached to it is a document entitled
7 Guidance on the Regulatory Requirements for
8 Criticality Analysis of Fuel Storage at Light-Water
9 Reactor Power Plants.
10 (Whereupon Exhibit Number 9
11 was marked for identification.)
12 Q. Do you recognize this document, Dr. Kopp?
13 A. Yes.
14 Q. To what extent, if any, did you participate in
15 the preparation of the guidance document that's attached
16 to this cover memo?
17 A. Well, I finalized it. It was begin maybe eight
18 or ten years previous to this by several members of our
19 branch. And we finally got it issued in 1998.
20 Q. So at the time this was written you approved it
21 and it represented your views.
22 A. Yes. It was an update of the previous guidance
23 that had gone out by the so-called Grimes letter, things
24 that had been approved since then. And we wanted to get
25 them all down in an official document and told what they

1 were.
2 Q. Does that document supersede the Grimes letter?
3 A. Well, it's not an additional NRC document.
4 It's a memo from me to my branch chief and it was put in
5 the PDR, but as such it is not an official NRC document.
6 Q. But the Grimes letter is in contrast?
7 A. Well, the Grimes letter would be equivalent to
8 a generic letter nowadays. I guess back in those times
9 they didn't have such a thing. So the Grimes letter went
10 through a series of compariances, various, I'm sure how
11 high. But this only went from myself to my branch chief.
12 But it presents no new policy, it's just an
13 update of existing methodologies and approvals that have
14 been made by the staff since the so-called Grimes letter.
15 Q. So this document summarizes the state of
16 existing regulatory guidance?
17 A. That's right.
18 Q. Would you please turn to page 7 of the attached
19 guidance document. Do you see towards bottom of the page
20 Section B entitled Additional Considerations?
21 A. Yes.
22 Q. Do you see also paragraph 2 under heading B?
23 A. Yes.
24 Q. Would you read paragraph 2 to yourself.
25 A. When this, there's a sentence in paragraph 2

1 that states, "Normally a misloading error involving only
2 a single assembly need be considered unless there are
3 circumstances that make multiple loading errors
4 creditible." Is that correct?
5 A. Yes.
6 Q. In evaluating a license amendment application
7 or a licensing application for spent fuel storage that
8 involves criticality analysis, does the NRC apply this
9 particular sentence, this consideration raised in this
10 sentence to the application?
11 A. It applies in that normally we consider, we
12 require an analysis of of a single misloading event.
13 Q. Normally does the staff make any determination
14 as to whether there are circumstances that's make
15 multiple loading errors credible?
16 A. The staff doesn't. No.
17 Q. Does the staff ask a licensee or license
18 applicant to do that?
19 A. Not that I'm aware of, no.
20 Q. Did the staff ask the license applicant to do
21 that in the Harris case?
22 A. No.
23 Q. So is it fair to say that the staff simply
24 doesn't apply the aspect of this sentence which says
25 "unless there are circumstances that make multiple

18

1 loading errors credible?"

2 MS. UTTAL: Objection. You can answer.

3 A. The reason this is in here is that if something

4 developes in the future, that I can't foresee now what

5 the circumstance would be, but we wanted something in

6 here to cover possible circumstances in the future where

7 more than a single misloading might be feasible.

8 We have not run into that as far as I know to

9 the present time, but to make this all encompassing

10 instead of revising it for the future, we decided to put

11 something like that to cover future possible

12 circumstances.

13 Q. But the purpose of this is to provide for such

14 consideration in case you should run into such

19

15 circumstances; is that right?

16 A. Yes.

17 Q. But you don't go looking for them, you just

18 wait to see if you run into them.

19 A. Yes.

20 Q. Please describe what circumstances, if any, and

21 under what regulatory requirements, if any, the NRC

22 requires recording of the misplacement of fresh or spent

23 fuel in spent fuel storage pools.

24 A. It would probably be a licensing, an LER. If

25 you violated your tec spec requirements or fuel loading

1 patterns there would be a license event report that is
2 required.
3 Q. Okay. And in an earlier discussion I had asked
4 you if you could give me a date when, I had asked you to
5 give me a date when burnup limits were first included in
6 tec specs, and what was the date you gave me?
7 A. I think it was early '80ies; 1982, 1983.
8 Q. So would that correspond to fuel loading
9 requirements?
10 A. I'm trying to think of when we went to
11 multi-region racks and whether it was, if it was only due
12 to burnup credit or whether there may have been some
13 other reason. I'm not sure, but I would think that would
14 be around the time we first --
15 Q. To your knowledge does the NRC keep records,
16 data, or documents that describe the practical experience
17 of nuclear power plant operators with fresh or spent fuel
18 misplacement in fuel storage pools?
19 A. Well, in the sense that these events are
20 reported to LER's, there would be a record of them.
21 Whether there's a compilation of them I'm not sure. But
22 they're certainly available.
23 Q. For your purposes or for your divisions
24 purposes of evaluating criticality analyses does the NRC
25 keep any such compilations?

1 A. Not that I know of. Mainly for the reason that
2 we require a misloading event to be analyzed anyway.
3 Q. I'm sorry, I didn't --
4 A. We require a misloading event to be one of the
5 analyzed accidents. Whether there are incidents in the
6 past or not does not seem to be of concern here since we
7 require the event anyway to be analyzed.
8 Q. Do you see any difference between misplacement,
9 misidentification and mischaracterization of spent fuel
10 assemblies, or fuel assemblies?
11 A. Would you please -- misplacement --
12 Q. Misplacement, misidentification and
13 mischaracterization.
14 A. It would seem to me it would be three separate
15 entities, items.
16 Q. Has the NRC ever evaluated whether a single
17 misidentification error could lead to multiple
18 misplacement of fuel?
19 A. No.
20 Q. Has the NRC ever evaluated whether a single
21 mischaracterization error could lead to multiple
22 misplacement of fuel?
23 A. No. When you say the "NRC," I'm answering for
24 myself. For my experience with this analysis. I don't
25 recall any other offices in the NRC that would be

1 evaluating this either. I can't speak for them.
2 Q. And are you familiar with any industry
3 evaluations of either of those two things, either
4 mischaracterization or misidentification leading to
5 misplacement of more than one assembly?
6 A. No.
7 Q. Has the NRC performed or obtained any analysis
8 or evaluation of the practical experience of nuclear
9 power plant operators with fresh or spent fuel
10 misplacement in fuel storage pools?
11 A. Obtained what?
12 Q. Do you want me to read it again?
13 A. Yes, please.
14 Q. Has the NRC performed or obtained any analysis
15 or evaluation of the practical experience of nuclear
16 power plants operators with fresh or spent fuel
17 misplacement in fuel storage pools?
18 MS. UTTAL: Objection. Could you be more
19 percise in your term about practical experience. Do
20 you mean an analysis of actual events?
21 Q. By practical experience I mean actual events.
22 A. The question was have we evaluated any actual
23 events of misloadings?
24 Q. Right. Well, have you evaluated the composite
25 experience of licensees with misloading events?

19

1 A. No.

2 Q. Has the NRC performed or obtained any analysis

3 of the probability of misplacing fresh or spent fuel in

4 fuel storage pools?

5 A. As I said before, we assume the probabilities

6 is a hundred percent because we require that analysis to

7 be performed. We require a misplaced fuel assembly to be

8 analyzed.

9 Q. But let me just clarify. You require as a

10 matter of practice the misplacement of a single fuel

11 assembly to be analyzed.

12 A. Right.

13 Q. My question was broader than that.

14 A. Could you repeat it.

15 Q. Has the NRC performed or obtained any analysis

20

16 of the probability of misplacing fresh or spent fuel in

17 fuel storage pools?

18 Why don't I try answering and you tell me if my

19 answer is correct.

20 A. Yeah. Throwing in the probabilities is what is

21 confusing me. As I said, by requiring the event to be

22 analyzed we assume its probability is a hundred percent

23 in a single misplaced assembly.

24 Q. And you haven't looked at the probability of

25 misplacing more than one fuel assembly.

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

Q. Could you explain that.

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

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

Q. Was this analysis documented anywhere?

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

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

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

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

A. Sure.

20

1 Q. Okay. You just described for me an analysis
2 that the staff did assuming that there were 80 misplaced
3 fuel assemblies in Pools C or D.
4 A. Yes.
5 Q. And that the boron was present --
6 A. Yes.
7 Q. -- at 2000 PPM.
8 A. Right.
9 Q. And you had told me earlier that at the other
10 extreme you have evaluated a situation where there's no
11 fuel misplacement but there's no boron.
12 A. That's right.
13 Q. And you've also evaluated, or Holtec has
14 evaluated, a situation where there is one fuel
15 misplacement and no boron.
16 A. That's right. Which is beyond what we normally
17 require for analysis.
18 Q. So is it correct to say that those are
19 basically two extremes?
20 A. I would say so, yes.
21 Q. Are there other combinations of events that you
22 could envision that could cause criticality; for
23 instance, misplacement of less, somewhere between zero
24 and 80, or one in 80 fuel assemblies, and some diminution
25 in boron concentration in the pool?

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

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

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

Q. Please answer the question.

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

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

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

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

A. Just to satisfy ourselves and Dr. Thompson.

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

A. No.

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

about a break.

MS. CURRAN: Okay. Sure.

(Whereupon a break was taken.).

THE DEPONENT: May I clarify something?

MS. CURRAN: Sure.

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

Q. Yes.

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

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

Q. When was it put in it PDR?

A. I guess August 19th, 1998.

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

A. No. Within the branch.

1 Q. Oh, within the branch --
2 A. Right.
3 Q. -- you solicited comments, but not from the
4 industry.
5 A. No. No.
6 Q. So you put it in the PDR with thoughts that you
7 might get comments from industry if you put it there or,
8 maybe I misunderstood you. When you said something about
9 getting comments, you meant from within the branch?
10 A. Within the branch.
11 Q. And not from industry.
12 A. Right.
13 Q. Okay. In Dr. Turner's deposition he mentioned
14 that he thought there might be a list of fuel
15 misplacement events kept by the the NRC. Do you recall
16 that?
17 A. As I said, there are LER's that come in, but I
18 don't know if there's a compilation of them or not.
19 Q. So you're not?
20 A. I'm not familiar with them.
21 Q. Okay. Has the NRC performed any analysis of
22 the consequence of misplacing one or more fresh or spent
23 fuel assemblies in fuel storage pools?
24 A. Yes.
25 Q. And where would that be found?

1 A. The analysis I mentioned that was done this
2 past week or so that we're preparing a memo on, where we
3 misplaced an entire rack with fresh fuel assemblies.
4 Q. All right. But that's the only one that you
5 know of?
6 A. That we did ourselves, that we analyzed
7 ourselves? Yes.
8 Q. Do you know of any that have been prepared by
9 any other entity?
10 MS. UTTAL: Are you speaking of CP&L's
11 application or any application?
12 Q. Anyone.
13 MS. UTTAL: Any kind of analysis of
14 misplacement.
15 A. You mean where there have been multiple
16 misplacements?
17 MS. UTTAL: She said single or more.
18 Q. Single or multiple.
19 A. I'm not certain. I have seen statements in the
20 past that to the effect that an entire rack could be
21 misloaded with fresh fuel assemblies and with credit for
22 boron, one would still maintain the five-percent
23 subcriticality margin. I have seen that with various
24 other submittals.
25 But we've never varified that ourselves. and

that's why we decided this week to actually do a calculation and see if would be true for Shearon Harris. And we found we are subcritical for the entire rack.

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

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

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

A. Right.

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

A. It would be the same as the LER's for fuel

1 misplacements. There would be the LER's as far as I
2 know. We don't compile them but they're available.
3 Q. Has the NRC performed or obtained any analysis
4 or evaluation of nuclear power plant operator's
5 experience with controlling boron concentrations in fuel
6 storage pools?
7 A. Not that I know of.
8 MS. CURRAN: I'd like to ask the court reporter
9 to mark as Exhibit 10 an October 25th, 1996 letter
10 from Timothy E. Collins, Acting Chief, Reactor
11 System Branch, Division of System Safety and
12 Analysis, NRC, to Mr. Tom Green, Chairman
13 Westinghouse Owner's Group. Subject: Acceptance
14 for Referencing of Licensing Topical Report
15 WCAP-14416-P, Westinghouse Special Fuel Rack
16 Criticality Analysis Methodology.
17 Attached to this cover letter is a Safety
18 Evaluation by the Office of Nuclear Reactor
19 Regulation relating to Topical Report WCAP-14416-P.
20 (Whereupon, Exhibit Number 10 was
21 marked for identification.)
22 Q. Dr. Kopp, are you familiar with this document?
23 A. Yes, I am.
24 Q. If you would turn to page 10 -- actually page
25 10 is a continuation of a discussion that starts on page

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

A. Yes.

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

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

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

For other plants, such as Shearon Harris, which do not take credit for soluble boron during normal conditions, the fact that they calculate the five percent

1 subcriticality margin in pure water takes care of the
2 boron dilution event, that is complete dilution.
3 For these newer plants that want to take credit
4 for the new methodology. They still must show they are
5 subcritical with no boron, k effective is less than one,
6 but to meet the k arc criteria, k effective less than or
7 equal to .95, they can take credit for a certain amount
8 of soluble boron. So because of that we require them now
9 to do a boron solution analysis to show that they would
10 get them below .95 dilution event.
11 Q. Okay. But Reference 29 in parentheses, when I
12 turn to the back of this SER, Reference 29 is "Cassidy,
13 B., et. al., Westinghouse Owners Group Evaluation of the
14 Potential for Diluting PWR Spent Fuel Pools, WCAP-14181,
15 July 1995."
16 How does that Reference 29 relate to what we
17 were just reading on page 10?
18 A. That was a companion to this Westinghouse
19 report which requested credit for partial boron. In
20 order to prove that methodology I said they have to do a
21 boron dilution event analysis. And this other report
22 that you referenced shows how to do an analysis of a
23 boron dilution event in the PWR.
24 Q. So the reason for the mention of Reference 29
25 is that this is a way for licensees to do the boron

1 dilution analysis and that, that will meet NRC approval?
2 A. When they want credit for this methodology,
3 partial boron credit, yes.
4 Q. And has the NRC approved Reference 29 for that
5 purpose?
6 A. No. The approval of a boron dilution event we
7 decide is done on a case by case basis because the plans
8 vary so much. The amount of, the volume of water that
9 can be inserted into a pool for dilution varies from
10 plant to plant through the mode of inserting it, the
11 capacity of the pools vary. We decided a generic
12 dilution event would not be worth anything or worth much,
13 so we decided to, the people that wanted to accept this
14 methodology for partial boron credit would have to do a
15 plan specific for boron dilution analysis for their
16 specific spent fuel pool. That's why that boron dilution
17 event was never approved or accepted. It was a generic
18 type of topical report.
19 Q. Okay.
20 Q. Has the NRC performed or obtained any analysis
21 of the probability and/or consequences of potential
22 accidents resulting from improper boron concentration in
23 fuel storage pool water?
24 A. Only the analysis that shows that the zero PPM
25 of boron when there's still a five-percent subcritical

complete dilution.

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

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

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

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

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

MR. HOLLAWAY: I have one question.

CROSS-EXAMINATION

BY MR. HOLLAWAY:

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

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

1 MR. HOLLAWAY: That's all. I have no other
2 questions.
3 MS. UTTAL: I have no questions.
4 MS. CURRAN: Okay.
5
6 * * * * *
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9 (Whereupon, these proceedings concluded at 3:30 p.m.)
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STATE OF FLORIDA)

COUNTY OF PINELLAS)

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

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

Dale E. DeFranco

Court Reporter

NOTARY PUBLIC STATE OF FLORIDA

Dale E. DeFranco
Notary Public, State of Florida
Commission No. CC 570095
My Commission Exp. 07/17/2000

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14

Laurence I. Kopp
Senior Reactor Engineer

Education

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

Employment

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

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

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

Federal Electric Corporation, Senior Programmer, 1957-1959

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

EXHIBIT

Professional Societies

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

Publications

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

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

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

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

1968.

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

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

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

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

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

DRAFT REGULATORY GUIDE AND VALUE/IMPACT STATEMENT

December 1981
Division 1
Task CE 913-5

Contact: C. Schulten (301)443-5910

PROPOSED REVISION 2* TO REGULATORY GUIDE 1.13

SPENT FUEL STORAGE FACILITY DESIGN BASIS

A. INTRODUCTION

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

B. DISCUSSION

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

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

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

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

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

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

Primary facility design objectives are:

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

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

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

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

1. LOSS OF WATER FROM STORAGE POOL

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

2. MECHANICAL DAMAGE TO FUEL

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

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

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

3. LIMITING POTENTIAL OFFSITE EXPOSURES

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

The staff considers Seismic Category I design assumptions acceptable for the spent fuel pool cooling, makeup, and cleanup systems. Tornado protectior requirements are acceptable for the water makeup source and its deliver ,ystem, the pool structure, the building housing the pool, and the filtratior .tilatior system. Regulatory Guide 1.52, "Design, Testing, and Maintenance Cr -ia for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System f ltration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," a egulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal V. .ilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear

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

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

C. REGULATORY POSITION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

D. IMPLEMENTATION

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

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

APPENDIX A

NUCLEAR CRITICALITY SAFETY

1. SCOPE OF NUCLEAR CRITICALITY SAFETY ANALYSIS

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

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

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

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

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

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

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

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

2. CALCULATION METHODS AND CODES

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

3. METHOD TO ESTABLISH SUBCRITICALITY

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

$$k_s \leq k_a$$

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

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

where

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

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

Δk_u = the uncertainty in the benchmark experiments, and

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

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

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

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

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

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

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

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

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

4. STORAGE RACK ANALYSIS ASSUMPTIONS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5. USE OF NEUTRON ABSORBERS IN STORAGE RACK DESIGN

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

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

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

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

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

(2) Periodically verify continued presence of absorber.

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

6. CREDIT FOR BURNUP IN STORAGE RACK DESIGN

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

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

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

DRAFT VALUE/IMPACT STATEMENT

1. PROPOSED ACTION

1.1 Description

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

1.2 Need for Proposed Action

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

1.3 Value/Impact of Proposed Action

1.3.1 NRC

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

1.3.2 Government Agencies

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

1.3.3 Industry

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

1.3.4 Public

No major impact on the public can be foreseen.

1.4 Decision on Proposed Action

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

2. TECHNICAL APPROACH

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

3. PROCEDURAL APPROACH

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

4. STATUTORY CONSIDERATIONS

4.1 NRC AUTHORITY

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

4.2 Need for NEPA Assessment

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

5. CONCLUSION

Regulatory Guide 1.13 should be updated.